Boston Edison
Pilgrim Nuclear Power Station
Rocky Hill Road
Plymouth, Massachusetts 02360

December 19, 1995
BECo Ltr. #95- 126

L. J. Olivier
Vice President Nuclear Operations
and Station Director

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

Docket No. 50-293
License No. DPR-35

Information Requested by German Federal Office for Radiation Protection

Please find enclosed the information relating to Pilgrim Nuclear Power Station requested by the German Federal Office for Radiation Protection. The request pertained to plant design relating to the ventilation system and emission data from 1973 to 1979. The plant design data is comprised of applicable sections of the plant Final Safety and Analysis Report and plant ventilation drawings. The emission data is comprised of applicable pages of the semi-annual reports made to the NRC during the period of concern. Additionally enclosed are copies of followup epidemiology studies that may be of interest to the German authorities. An index to the enclosures follows.

Please contact Jeffrey Calfa at (508) 830-8108 for questions relating to the plant design data and this letter in general and Thomas Sowdon at (508) 830-8834 for questions relating to the emissions data and the epidemiology followup studies. Thank you.

L.J. Olivier

L.J. Olivier

JPC/dmc/german

cc: Mr. R. Eaton, Project Manager
Division of Reactor Projects - I/II
Mail Stop: 14D1
U.S. Nuclear Regulatory Commission
1 White Flint North
11555 Rockville Pike
Rockville, MD 20852

U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406

JE25 1/1

Senior Resident Inspector
Pilgrim Nuclear Power Station

Drawings located in Central Files

Index of Information Provided

Item	Description
FSAR Section 5.1	Containment Summary
FSAR Section 5.2	Primary Containment System
FSAR Section 5.3	Secondary Containment System
FSAR Section 9.4	Gaseous Radwaste System
FSAR Section 10.9	Heating, Ventilation, and Air Conditioning
Drawing M210	Main Condenser Air Ejector and Offgas System
Drawing M254	Augmented Offgas System
Drawing M283	Secondary Containment Isolation Control Drawing
Drawing M287	Plant Ventilation Drawing
Drawing M288	Turbine Building Air Flow Diagram
Drawing M289	Reactor Building Air Flow Diagram
Drawing M290	Radwaste Area Air Flow Diagram
Drawing M294	Standby Gas Treatment System Control Diagram
Pilgrim Semi-Annual Radioactive Effluent Release Reports (applicable pages) from July 1972 to June 1979	Applicable pages demonstrate gaseous effluent releases from the plant during the applicable time frame
Review of the Southeastern Massachusetts Health Study, dated October 1992	Follow-up epidemiology study
Plausibility of the Results from the Southeastern Massachusetts Health Study	Follow-up epidemiology study

SECTION 5

CONTAINMENT

5.1 SUMMARY DESCRIPTION

5.1.1 General

The containment systems of Pilgrim Nuclear Power Station utilize a "multibarrier" concept which consists of two systems. The Primary Containment System (PCS) is a pressure suppression system which forms the first barrier. The Secondary Containment System (SCS) is a system which minimizes the ground level release of airborne radioactive materials, and forms the second barrier. The fuel, fuel cladding, and Reactor Primary System (RPS) form additional barriers to the release of fission products and are described in Section 3.2.

5.1.2 Primary Containment System

The PCS houses the reactor vessel, the Reactor Coolant Recirculation System and other branch connections of the Reactor Coolant System (RCS). The primary containment is a pressure suppression system consisting of a drywell, pressure suppression chamber which stores a large volume of water, a connecting vent system between the drywell and water pool, isolation valves, vacuum relief system, containment cooling systems, and other service equipment. The drywell is a steel pressure vessel in the shape of a light bulb, and the pressure suppression chamber is a torus shaped steel pressure vessel located below and encircling the drywell.

The PCS is designed to withstand the forces from any size breach of the nuclear system primary barrier up to and including an instantaneous circumferential break of the reactor recirculation piping, and provides a holdup time for decay of any radioactive material released. The PCS also stores sufficient water to condense the steam released as a result of a breach in the nuclear system primary barrier and to supply the Core Standby Cooling Systems (CSCS).

5.1.3 Secondary Containment System

The SCS encloses the PCS, the refueling and reactor servicing areas, new and spent fuel storage facilities, and other reactor auxiliary systems. The SCS serves as the only containment during reactor refueling and maintenance operations, when the primary containment is open, and as an additional barrier when the PCS is functional. The SCS consists of the reactor building, Standby Gas Treatment System (SGTS), main stack, Reactor Building Isolation and Control System (RBICS), and other service equipment.

The SCS is designed to withstand the maximum postulated seismic event, and be capable of providing holdup treatment, and an elevated release point for any fission products released tc it. In addition, the Reactor Building is designed to provide protection for the engineered safeguards and nuclear safety systems located in the building from all postulated environmental events including tornadoes.

5.2 PRIMARY CONTAINMENT SYSTEM

5.2.1 Safety Objective

The safety objective of the Primary Containment System (PCS) is to provide the capability in conjunction with other safeguard features, to limit the release of fission products in the event of a postulated design basis accident (DBA) so that offsite doses would not exceed the guideline values set forth in 10CFR100.

5.2.2 Safety Design Basis

1. The PCS shall have the capability of withstanding the conditions which could result from any of the postulated DBAs for which the PCS is assumed to be functional, including the largest amount of energy release and mass flow associated with the accident.

2. The PCS shall have a margin for metal water reactions and other chemical reactions subsequent to any postulated DBA for which the PCS is assumed to be functional, consistent with the performance objectives of the nuclear safety systems and engineered safeguards.

3. The PCS shall have the capability to maintain its functional integrity during any postulated external or environmental event.

4. The PCS shall have the capability to be filled with water as an accident recovery method for any postulated DBA in which a breach of the nuclear system primary barrier cannot be sealed.

5. The PCS, in conjunction with other Nuclear Safety Systems and engineered safeguards, shall have the capability to limit leakage during any of the postulated DBAs for which it is assumed to be functional, such that offsite doses do not exceed the guideline values set forth in 10CFR100.

6. The PCS shall have the capability to rapidly isolate all pipes or ducts necessary to establish the primary containment barrier.

7. The PCS shall have the capability to store sufficient water to supply the Core Standby Cooling System (CSCS) requirements.

8. The primary containment shall have the capability to be maintained during normal operation within the range of initial conditions assumed in the Station Safety Analysis in Section 14.

5.2.3 Description

5.2.3.1 General

The design employs a Low Leakage Pressure Suppression Containment System which houses the reactor vessel, the reactor coolant recirculating loops, and other branch connections of the Reactor Primary System. The Pressure Suppression System consists of a drywell, a pressure suppression chamber (torus) which stores a large volume of water, a connecting vent system between the drywell and the pressure suppression pool, isolation valves, Vacuum Relief System, Containment Cooling Systems, and other service equipment. The PCS design parameters are given on Table 5.2-1.

In the event of a Process System piping failure within the drywell, reactor water and steam will be released into the drywell gas space. The resulting increased drywell pressure forces a mixture of air, steam, and water through the vent system into the pressure suppression pool. The steam condenses rapidly in the suppression pool resulting in rapid pressure reduction in the drywell. Air transferred during reactor blowdown to the suppression chamber pressurizes the chamber, and subsequently is vented to the drywell through the vacuum relief system as the pressure in the drywell drops below that in the suppression chamber.

Cooling systems are provided to remove heat from the water in the suppression chamber. This provides for continuous cooling of the primary containment under the postulated DBA conditions for which the PCS is assumed to be functional. Isolation valves are provided to ensure containment of radioactive materials within the primary containment, which might be released from the reactor to the containment during the course of an accident. Other service equipment is provided to maintain the containment within its design parameters during normal operation.

The drywell (primary containment) coolers are designed to maintain drywell atmosphere temperatures within an acceptable range during normal station operation. See Table 5.2-2. The reduction of atmosphere temperature by the coolers will also result in partial condensation of water vapor when the incoming humidity levels are high.

The drywell fan motors are rated for continuous operation in atmospheres having 100 percent rh and 104°F temperatures. In the design of the cooler, the motor has been placed in the exhaust of the cooler where the leaving air temperature is a maximum of 95°F, so that the motor is exposed to the lowest humidity and lowest temperature atmosphere available within the drywell. Pressure increases to the 2.5 psig high drywell pressure condition used to sense a possible loss of coolant accident (LOCA) would not affect the continued operability of the coolers. The drywell coolers are automatically shut down in the event of a LOCA combined with the loss of offsite ac power.

The drywell coolers, including the fans, with their power and control systems were tested during the preoperational tests at the station to demonstrate the required operability of the power and control systems, the fans, and the Reactor Building closed cooling water supply to the coolers. The capability of the coolers to maintain the required drywell atmosphere temperatures was verified during the startup program as the drywell heat loads increased during the heatup and pressurization of the Nuclear Steam Supply System.

The Reactor Building Closed Cooling Water System (RBCCWS) piping supplying the drywell coolers will be revised to seismic Class I to maintain the pressure boundary integrity of this piping under seismic loading. Refer to Section 10.5.5.1. The drywell coolers were originally purchased as seismic Class I equipment to serve as pressure boundary only.

The PCS design loading considerations are given in Section 12 and Appendix C. The Station Safety Analysis presented in Section 14 demonstrates the effectiveness of the PCS as a radiological barrier. In addition, primary containment pressure and temperature transients from postulated DBAs are also presented in Section 14.

5.2.3.2 Drywell

The drywell is a steel pressure vessel with a spherical lower portion, 64 ft in diameter, and a cylindrical upper portion 34 ft 2 inches in diameter. The overall height is approximately 110 ft. The design, fabrication, inspection, and testing of the drywell vessel complies with requirements of the ASME Boiler & Pressure Vessel Code, Section III, Subsection B, Requirements for Class B Vessels, which pertain to containment vessels for nuclear power stations.

The drywell structure is designed for an internal pressure of 56 psig coincident with a temperature of 281°F with applicable dead, live, and seismic loads imposed on the shell. Thus, in accordance with the ASME Code, Section III, Code Case N-1312-(2), the maximum drywell pressure is 62 psig. Thermal stresses in the steel shell due to temperature gradients are taken into account in the design.

Special precautions not required by codes were taken in the fabrication of the steel drywell shell. Charpy V-notch specimens were used for impact testing of plate and forging material to give assurance of proper material properties. Plates, forgings, and pipe associated with the drywell have an initial NDT temperature of 0°F or lower when tested in accordance with the appropriate code for the materials. It is intended that the drywell will not be pressurized or subjected to substantial stress at temperatures below 30°F.

The drywell is enclosed in reinforced concrete for shielding purposes, and to provide additional resistance to deformation and buckling in areas where the concrete backs up the steel shell. Above the transition zone, the drywell is separated from the reinforced concrete by a gap of approximately 2 in. Shielding over the top of the drywell is provided by removable, segmented, reinforced concrete shield plugs.

In addition to the drywell head, one double door air lock and two bolted equipment hatches are provided for access to the drywell. The locking mechanisms on each air lock door are designed so that a tight seal will be maintained when the doors are subjected to design pressures. The doors are mechanically interlocked so that neither door may be operated unless the other door is closed and locked. The drywell head and equipment hatch covers are bolted in place and sealed with gaskets.

The spectrum of primary system leak rates up to a double ended blowdown of a recirculation line has been analyzed relative to the temperature and pressure response of the drywell. Steam issuing from a leak and expanding at constant enthalpy may result in a superheated containment atmosphere. The maximum amount of superheat possible is a function of both the source pressure (reactor pressure) and the receiver pressure (drywell). The enthalpy of saturated steam goes through a maximum value at a reactor pressure of 400 to 500 psia. Steam issuing from a leak at this pressure will result in the maximum superheat for a given containment pressure.

If a steam leak occurs, the containment pressure and temperature increase at a rate dependent on the size of the leak, containment characteristics, and the pressure of the reactor. The containment pressure and temperature rises as noncondensable gases are swept into the suppression chamber. Containment pressure levels off after all noncondensable gases are driven into the suppression chamber. The containment shell temperature rises as steam condenses on the relatively cool wall. When the drywell shell temperature reaches the saturation temperature dictated by this containment pressure, steam condensation is terminated. The only energy available to further increase the wall temperature is the superheat energy. The result is a decrease in the rate of temperature rise of the drywell shell and an increase in the bulk atmosphere temperature of the drywell.

Figure 5.2-1 illustrates the reactor vessel pressure response to steam leaks ranging in size from 0.02 to 0.50 ft^2. Figures 5.2-2 through 5.2-6 illustrate the containment response to steam leaks covering the same size range. The response of the containment to small steam leaks is slow, but the continued high reactor pressure results in high containment temperature, given enough time. Leaks so small that the high drywell pressure trip does not occur will not result in a high temperature. Leaks large enough to result in a high containment temperature will be large enough to sweep air into the suppression chamber and result in significant drywell pressure increase. Large leaks will either depressurize the reactor rapidly or result in auto-relief such that steam temperatures, reaching levels up to 330°F, will not persist long enough to result in structural wall temperatures exceeding 281°F.

Safety grade temperature monitoring instrumentation is provided in the Control Room so that activation of one of the two containment sprays would be effective in terminating the temperature rise because the superheat is quickly removed. The spray nozzles are designed to give a small particle size, and the heat transfer to the subcooled spray is very effective. Since the total amount of heat in the drywell atmosphere is low relative to the spray rate, the containment atmosphere temperature is quickly reduced to near the spray temperature.

A drywell pressure condition exceeding 10 psig was selected as the basis for determining when to initiate the containment spray. See Figure 5.2-7 for time required to reach 10 psig. The operator will be instructed to initiate the containment sprays if containment pressure exceeds 10 psig for longer than 30 min. Safety grade pressure monitoring instrumentation is provided in the Control Room for this purpose. This procedure will ensure that the containment wall never exceeds 281°F. Depressurization of the reactor vessel can take place at the normal rate, but depressurization is not required to ensure that the wall temperature remains below 281°F. The environmental conditions considered in the design of the reactor protective system instrumentation, engineered safety feature equipment, and the qualification tests that have been conducted are described in Section 7.1.8. The analyses of steam leaks inside the containment is given in detail in references 7 and 8.

5.2.3.3 Pressure Suppression Chamber and Vent System

5.2.3.3.1 General

The pressure suppression pool, which is contained in the pressure suppression chamber, initially serves as the heat sink for any postulated transient or accident condition in which the normal heat sink, main condenser, or Shutdown Cooling System is unavailable. Energy is transferred to the pressure suppression pool by either the discharge piping from the reactor pressure relief valves or the Drywell Vent System. The relief valve discharge piping is used as the energy transfer path for any condition which requires the operation of the relief valves. The Drywell Vent System is the energy transfer path for all energy releases to the drywell.

Of all the postulated transient and accident conditions, the instantaneous circumferential rupture of the reactor coolant recirculation piping represents the most rapid energy addition to the pool. For this accident the vent system, which connects the drywell and suppression chamber, conducts flow from the drywell to the suppression chamber without excessive resistance and distributes this flow effectively and uniformly in the pool. The pressure suppression pool receives this flow, condenses the steam portion of this flow, and releases the noncondensable gases and any fission products to the pressure suppression chamber air space.

5.2.3.3.2 Pressure Suppression Chamber

The pressure suppression chamber is a steel pressure vessel in the shape of a torus below and encircling the drywell, with a centerline vertical dia of 29 ft 6 in and a horizontal dia of 131 ft 6 in. The pressure suppression chamber contains approximately 84,000 ft^3 of water and has a net air space above the water pool of approximately 120,000 ft^3. The suppression chamber will transmit seismic loading to the reinforced concrete foundation slab of the Reactor Building. Space is provided outside of the chamber for inspection.

The toroidal suppression chamber is designed to the same material and code requirements as the steel drywell vessel. The material has an NDT temperature of 0°F or less.

5.2.3.3.3 Vent System

Large vent pipes connect the drywell and the pressure suppression chamber. A total of eight circular vent pipes are provided, each having a dia of 6.75 ft. The vent pipes are designed for the same pressure and temperature conditions as the drywell and suppression chamber. Jet deflectors are provided in the drywell at the entrance of each vent pipe to prevent possible damage to the vent pipes from jet forces which might accompany a pipe break in the drywell. The vent pipes are fabricated of SA-516 steel, and comply with requirements of the ASME Boiler and Pressure Vessel Code, Section III, Subsection B. The vent pipes are provided with expansion joints which are enclosed within sleeves, to accommodate differential motion between the drywell and suppression chamber.

The drywell vents are connected to a 4 ft 9 in dia vent header in the form of a torus which is contained within the airspace of the suppression chamber. Projecting downward from the header are 96 downcomer pipes, 24 inches in dia, terminating approximately 3.00 to 3.25 ft below the water surface of the pool. The vent header has the same temperature and pressure design requirements as the vent pipes. Vent pipes and vent headers are braced to withstand expected loads from steam blowdown into the pool.

5.2.3.3.4 Pressure Suppression Pool

The pressure suppression pool is approximately 84,000 ft^3 of demineralized water contained within the pressure suppression chamber. It serves both as a heat sink for postulated transients and accidents and as a source of water for the CSCS.

The suppression pool receives energy in the form of steam and water from the reactor pressure relief valve discharge piping, or the drywell vent system downcomers which discharge under water. The steam is condensed by the suppression pool. The condensed steam and any water carryover cause an increase in pool volume and temperature. Energy can be removed from the suppression pool when the Residual Heat Removal System (RHRS) is operating in the suppression pool cooling mode.

The suppression pool is the primary source of water for the Core Spray and Low Pressure Coolant Injection (LPCI) Systems, and the secondary source of water for the Reactor Core Isolation Cooling (RCIC) and High Pressure Coolant Injection (HPCI) Systems. The water level and temperature of the suppression pool are continuously monitored in the main control room.

5.2.3.4 Penetrations

5.2.3.4.1 General

Containment penetrations have the following design characteristics:

1. They are designed for the same pressure and temperature conditions as the drywell and pressure suppression chamber

2. They are capable of withstanding the forces caused by impingement of the fluid from the rupture of the largest local pipe or connection without failure

3. They are capable of accommodating the thermal and mechanical stresses, which may be encountered during all modes of operation including environmental events, without failure

4. They are capable of withstanding the maximum reaction that the pipe to which they are attached is capable of exerting

The penetration schedule, including the number and size of these penetrations, is shown on Table 5.2-3. Load combinations and allowable stresses are described in Appendix C.

5.2.3.4.2 Pipe Penetrations

Two general types of pipe penetrations are provided. Type 1 is used where the design must accommodate thermal movement. Figure 5.2-9 is typical of this type of penetration. Type 2 is used where stresses due to thermal movement are relatively small. Typical penetrations of this type are illustrated on Figures 5.2-10 and 5.2-11. Figure 5.2-12 shows a typical instrument penetration.

The piping penetrations which have special provisions for thermal movement, such as the steam lines, are shown on Figure 5.2-9. In these penetrations, the process line is enclosed in a guard pipe that is attached to the main steam line through a multiple head fitting. This fitting is a one-piece forging with integral flues or nozzles and is designed to meet all requirements of the ASME Boiler and Pressure Vessel Code, Section III, Class B. The forging is radiographed and ultrasonically tested as specified by this code. The guard pipe and flued head are designed to the same pressure requirements as the process line. The process line penetration sleeve is welded to the drywell, and extends through the biological shield where it is welded to a two-ply expansion bellows assembly, which in turn is welded to the flued head fitting. The pipe is guided through pipe supports at the end of the penetration assembly to allow steam pipe movement parallel to the penetration, and to limit pipe reactions of the penetration to allowable stress levels.

Where necessary, the penetration assemblies are anchored outside the containment to limit the movement of the line relative to the containment. The bellows accommodates the relative movement between the pipe and the containment shell.

The design of the penetration takes into account the stresses associated with normal thermal expansion, live and dead loads, seismic loads, and loads associated with a LOCA within the drywell. The design takes into account the loadings given above in addition to the jet force loadings resulting from any pipe failure. The resultant stresses in the pipe and penetration for the condition do not exceed 90 percent of the material yield stress.

The cold piping, ventilation duct, and instrument line penetrations are generally welded directly to the sleeves. In some cases, where stress analyses indicate the need, double flued head fittings are used. Bellows and guard pipes are not necessary in these designs, since the thermal stresses are small and are accounted for in the design of the weld joint.

5.2.3.4.3 Electrical Penetrations

The electrical penetrations include electrical power, signal, and instrument leads. Typical electrical penetrations are shown on Figures 5.2-13, 5.2-14, and 5.2-15. The penetrating sleeve is welded to the primary containment vessel. Medium voltage power penetrations primary seals are made of alumina-ceramic materials. The seals are formed at 1,300°F or higher, and thus the temperatures to which the seals would be exposed during a LOCA would have no adverse effect on their leaktightness characteristics.

The electrical penetrations used for low voltage power, control, and instrumentation cable and for coaxial cable utilize either Al_2O_3 or a bonding resin to maintain the leaktight integrity of the containment penetrating sleeves. A prototype of the penetration assembly which utilizes a bonding resin has been tested by exposing the interior face of the penetration assembly to the following environmental conditions: 281°F, 63 psig internal pressure, 90-100 percent rh for 10 days. An additional test at 320°F, 125 psig internal pressure and 90-100 percent rh for 2 hr was conducted. The pressure retaining capability of the penetration assembly was maintained throughout the duration of the tests.

The leak rate was monitored during the test and did not exceed 24 cc/hr of nitrogen through the inner seal. The outer seal is not exposed to high temperatures during an accident and therefore the overall leak rate through both seals is 10^{-6} cc/sec.

Additional tests were planned to certify the pressure retaining capability of those penetrations utilizing bonding resin at 340°F, 100 percent rh for 30 min.

A prototype of the penetrations using a polysulfone seal has been qualified to the following environmental conditions: 340°F, 110 psig for 6 hr; 320°F, 75 psig for 3 hr; and 260°F, 20 psig for 12 days. The inboard and/or outboard seal possessed a leak rate that was less than 5.3×10^{6} cc/sec helium.

Section 7.1.8 states that qualification tests were to be conducted on the medium voltage electrical penetrations, including leakage tests, following environmental exposures in excess of the design basis LOCA conditions.

5.2.3.4.4 Traversing Incore Probe Penetrations

Traversing incore probe (TIP) guide tubes pass from the Reactor Building through the primary containment. Penetration of the guide tubes through the primary containment are sealed by means of brazing which meets the requirements of the ASME Boiler and Pressure Vessel Code, Section VIII. These seals would also meet the intent of Section III of the code even though the code has no provisions for qualifying the procedures or performances.

5.2.3.4.5 Personnel and Equipment Access Locks

One personnel access lock is provided for access to the drywell. The lock has two gasketed doors in series, and each door is designed to withstand the drywell design pressure. The doors are mecnanically interlocked to ensure that at least one door is locked at all times when primary containment is required. The locking mechanisms are designed so that a tight seal will be maintained when the doors are subjected to either the design internal or external pressure. The seals on this access opening are capable of being tested for leakage.

A personnel access hatch with testable seals is provided on the drywell head. This hatch is bolted in place.

Two equipment access hatches with testable seals are also provided. These hatches are bolted in place.

5.2.3.4.6 Access to the Pressure Suppression Chamber

Access to the pressure suppression chamber is provided at two locations from the Reactor Building. There are two 4 ft dia manhole entrances with double gasketed bolted covers connected to the chamber by 4 ft dia steel pipes.

5.2.3.4.7 Access for Refueling Operations

The drywell vessel head is removed during refueling operations. The head is held in place by bolts and is sealed with a double-seal arrangement.

5.2.3.5 Primary Containment Isolation Valves

5.2.3.5.1 General Criteria

The basic function of all primary containment isolation valves is to provide necessary isolation to the containment in the event of accidents or similar critical conditions when the free release of containment atmosphere cannot be permitted. The containment isolation valves are listed on Table 5.2-4. This table also defines the valve status (normally open or normally closed) during normal reactor operation and shows the signals required to initiate their desired operation. The primary containment isolation valves are grouped into four basic classes.

Class A valves are on process lines that communicate directly with the reactor vessel and penetrate the primary containment. These lines require two valves in series, one inside the primary containment and one outside the primary containment. They are located as close to the primary containment boundary as practical. Except in the case of check valves, both valves shall close automatically on isolation signal. Both valves shall receive the isolation (closure) signal even if normally closed during reactor operation. Since check valves close on reverse process flow, they are used to isolate some incoming lines. All Class A valves except check valves are capable of remote manual control from the control room.

Class B valves are on process lines that do not directly communicate with the reactor vessel, but penetrate the primary containment and communicate with the primary containment free space. These lines require two valves, in series, both of them located outside the primary containment, and as close to the primary containment boundary as practical. Except in the case of check valves, both valves close automatically on isolation signal. Both valves receive the isolation closure signal even if normally closed during reactor operation. See Table 5.2-4 for valve status during reactor operation. All Class B valves except check valves are capable of remote manual control from the control room.

Class C valves are on process lines that penetrate the primary containment but do not communicate directly with the reactor vessel, with the primary containment free space, or with the environs. Class C lines require only one valve which closes automatically by process action (i.e., reverse flow) or by remote manual operation from the control room (Reference 6, Section 5.2.9).

Motive power for the valves on process lines which require two valves shall be from physically independent sources to provide a high probability that no single accidental event could interrupt motive power to both closure devices.

Variations to the above definitions are referenced on Table 5.2-4 by their class designations followed by an "X" suffix. The lines in this class are generally instrument lines or lines used for core cooling.

Automatic isolation valves, in the usual sense, are not used on the inlet lines of the Reactor Core and Containment Cooling Systems and Reactor Feedwater Systems, since operation of these systems is essential following a design basis LOCA. Since normal flow of water in these systems is inward to the reactor vessel or to the primary containment, check valves located in these lines will provide automatic isolation, if necessary.

No automatic isolation valves are provided on the Control Rod Drive System hydraulic lines. These lines are isolated by the normally closed hydraulic system control valves located in the Reactor Building, and by check valves comprising a part of the drive mechanisms.

TIP lines and small diameter instrument lines are not provided with automatic isolation valves.

5.2.3.5.2 Additional Considerations

Effluent lines such as main steam lines, which connect to the reactor vessel or which are open to the primary containment, have air-powered valves. This arrangement provides a high reliability with respect to functional performance. These valves are closed automatically by the signals indicated on Table 5.2-4.

The MSIV's are also connected to the nitrogen supply system. This redundant source of MSIV actuation results in greater system reliability.

TIP system guide tubes are provided with an isolation valve which closes automatically upon receipt of proper signal and after the TIP cable and fission chamber have been retracted. In series with this isolation valve, an additional or backup isolation shear valve is included. Both valves are located outside the drywell. The function of the shear valve is to assure integrity of the containment in the unlikely event that the other isolation valve should fail to close or the chamber drive cable should fail to retract if it should be extended in the guide tube during the time that containment isolation is required. This valve is designed to shear the cable and seal the guide tube upon an actuation signal. Valve position (full open or full closed) of the automatic closing valves will be indicated in the control room. Each shear valve will be operated independently. The valve is an explosive type valve and each actuating circuit is monitored. In the event of a containment isolation signal, the TIP system receives a command to retract the traveling probes. Upon full retraction, the isolation valves are then closed automatically. If a traveling probe were jammed in the tube run such that it could not be retracted, instruments would supply this information to the operator, who would in turn investigate to determine if the shear valve should be operated.

The two 18 in purge and vent line pipe entrances into the drywell have been provided with baffle plates to prevent debris from entering the lines during an accident. Any debris would threaten the ability to close the applicable isolation valve.

The N_2 makeup and vent isolation valves are used to relieve high drywell pressure during nonaccident conditions. However, these valves may be used after an accident provided the required power supplies are available and a low-low water level signal is not present. Section 5.4.3 describes the N_2 makeup and vent valves used following an accident condition.

Lines, such as those of the RBCCWS which do not connect to the Reactor Primary System or open into the primary containment, are provided with at least one ac-powered valve on the effluent line and a check valve on the influent line.

The Control Rod Hydraulic System is provided with three valves which can be utilized for isolation purposes. The first is a ball check valve which comprises an internal portion of the control drive mechanism. The other valves are normally closed hydraulic system control valves located in the Reactor Building.

5.2.3.5.3 Instrument Piping Connected to the Reactor Primary System

Instrumentation piping connecting to the Reactor Primary System which leaves the primary containment is dead-ended at instruments located in the Reactor Building. These lines are provided with flow limiting orifices, manual isolation valves, and excess flow check valves.

Instrument sensing lines that originate within the reactor coolant pressure boundary and penetrate the primary containment are 1 in dia seismic Class I lines; 1/4 in dia orifices are installed in each of these lines inside the primary containment. This orifice size was selected to provide the same effective fluid cross sectional area as the excess flowcheck valves when fully open. A manually operated stop valve and excess flowcheck valve are installed in each line immediately outside, and as close as practicable to the primary containment consistent with the requirement for access to the stop valve. The combination of orifice and excess flowcheck valve will reduce leakage to as low a value as practicable in the unlikely event of line failure. A line failure downstream of the excess flow check valve will result in a maximum leakage rate of 2 gpm prior to actuation. A failure of the excess flowcheck valve body or the instrument line upstream of this valve would result in a maximum leakage rate of 20 gal/min. In each of these instances the leakage is well within the capability of the Reactor Coolant Makeup System.

The amount of steam released to the Reactor Building from a 20 gal/min leak would not result in a failure of secondary containment. If the Reactor Building is not isolated, there would not be any significant pressure rise due to the relatively high Reactor Building ventilation exhaust rates. If the Reactor Building is isolated, the operation of one standby gas treatment filter train will prevent Reactor Building pressure from exceeding its design value.

An analysis of the potential offsite exposure that would result from a 20 gal/min leak into the Reactor Building has been performed. Such a leak corresponds to an assumed failure of an instrument line outside the primary containment but upstream of the excess flowcheck valve. It was assumed in the analysis that manual shutdown and depressurization would be initiated within 30 min. The delay of

30 min is extremely conservative considering the numerous ways such a leak may be detected.

The analysis assumed that steam from the leak would be released to the environment through the normal ventilation path until the reactor had been depressurized. Based on these assumptions, the total dose at the site boundary for the duration of exposure was computed to be 0.15 rem to the thyroid, which is substantially below the guidelines of 10CFR100.

Pressure retaining welds of instrument sensing lines that are part of the reactor coolant pressure boundary receive magnetic particle or liquid penetrant examination of the last pass.

Instrument line "bundles" are routed so as to minimize the potential for accidental damage. They are generally routed high in compartments to ensure they are not stepped on or otherwise damaged. The lines are equipped with flow limiting orifices and excess flowcheck valves and are of the same size and schedule; therefore, the possibility of one line causing failure in another is extremely remote.

The containment penetrations for these sensing lines are shown on Figure 5.2-12. The 10 in drywell penetration sleeve contains six, equally spaced, 1 in, schedule 80 stainless steel instrument lines. The manual isolation valves are 1 in stainless steel globe valves and are located as close to the penetration as practical, consistent with the need for access to the valve. The excess flowcheck valves close automatically on flow in excess of 2 gal/min. Neither the manual isolation valves nor the excess flowcheck valves are equipped with position indicators. Regular monitoring of measured variables and comparison between redundant instruments provides operating personnel with sufficient information to identify malfunctioning or inoperative instruments and sensing lines. Operating and/or testing procedures will assure the operability of the safety related instrument lines and their associated orifices and excess flow check flows.

An analysis was conducted to determine the amount of Reactor Building ventilation that would be required to prevent exceeding the design internal pressure of the Reactor Building for an instrument line blowdown through a 1/4 in orifice. The required ventilation flow rate under these conditions is approximately 2,000 ft^3/min, which is far below the available flow rate through either the normal Reactor Building Ventilation System or the Standby Gas Treatment System (SGTS). An instrument line failure will therefore not result in a loss of integrity of the Secondary Containment System (SCS).

An estimate of the potential offsite exposure that would result from an instrument line failure has been calculated. The assumptions employed in this analysis were:

1. An instrument line failure occurs and results in an initial blowdown of 2.2 lb mass/sec into the Reactor Building

2. This blowdown continues undiminished and undetected for a period of 30 min

3. After a period of 30 min, the reactor is shut down, depressurized, and cooled down at a controlled 100°F/hr

4. The water which flashes to steam is carried out of the Reactor Building by the normal ventilation system for the duration of the blowdown

5. The I-131 concentration in the blowdown is 6.1×10^{-2} microcurie/ml and the total iodine concentration is 1.6×10^{0} microcurie/ml

6. The atmospheric diffusion factor (X/O) for a ground level release, 500 m distance to site boundary, and wake dilution factor of 3 is 5×10^{-4} sec/m^3

7. The breathing rate is 3.47×10^{-4} m^3/sec

The above estimates assume that corrective action would not begin for a period of 30 min. The detection of a sensing line break would be almost immediate by one or a combination of the means listed below. Proper corrective action would then be taken by the operating staff in accordance with station procedures such that the leak would be isolated or station shutdown and depressurization be initiated. It is believed that it is not credible to assume no operator action would be taken in 30 min to terminate the consequences, and that the analysis based on a 30 min allowance for these actions is very conservative.

Sensing line break detection means are:

1. By a scram, annunciation, and possible instrument readouts and/or initiation of reactor safeguards systems if rupture occurred on a Reactor Protection System instrument line

2. By annunciation of the control function, either high or low in the control room

3. Operator comparing readings with several instruments monitoring the same process variable such as reactor level, jet pump flow, and steam pressure

4. By increases in area temperature monitor readings and high temperature alarms in the Reactor Building, and/or ventilation exhaust air ducts

5. By a general increase in the area radiation monitor readings throughout the Reactor Building

6. The leak should be audible either inside the Turbine Building or outside the Reactor Building to the operating staff members on a normal tour

7. By detecting the leak as soon as an access door to the Reactor Building is opened or approached

Routine surveillance and the multiplicity of detection methods on the part of the operator as given in items 1 through 7 above, represent an adequate means for detection of incipient or sudden failure of these small diameter instrument lines and components.

5.2.3.6 Venting and Vacuum Relief System

1. General

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber and reactor building so that the structural integrity of the containment is maintained. The vacuum relief system from the pressure suppression chamber to reactor building consists of two 100-percent vacuum relief breakers (2 parallel sets of 2 valves in series). Operation of either system will maintain the pressure differential less than 2.0 psig: the external design pressure. One valve may be out of service for repairs for a period of 7 days. If repairs cannot be completed within 7 days, the reactor coolant system is brought to a condition where vacuum relief is no longer required. |

The capacity of the 10 drywell vacuum relief valves are sized to limit the pressure differential between the suppression chamber and drywell during post-accident drywell coolant operations to the design limit of 2.0 psig. They are sized on the basis of the Bodega Bay pressure suppression system tests[1]. The ASME Boiler and Pressure Vessel Code, Section III, Subsection B, for this vessel allows a 5 psig vacuum; therefore, with two vacuum relief valves secured in the closed position and eight operable valves, containment integrity is not impaired. |

Reactor operation is permissible if the bypass area between the primary containment drywell and suppression chamber does not exceed an allowable area. The allowable bypass area is based upon analysis considering primary system break area, suppression chamber effectiveness, and containment design pressure. Analyses show that the maximum allowable bypass area is 0.2 ft^2.

Reactor operation is not permitted if the differential pressure decay rate is demonstrated to exceed 25 percent of allowable, thus providing a margin of safety for the primary containment in the event of a small break in the primary system.

2. Relief Valve Monitors

The drywell to torus vacuum breakers are installed to assure that the drywell pressure is at least equal to or greater than the pressure in the torus. In addition, when the vacuum breakers are in the closed position, the drywell atmosphere (postulated steam) is directed through the suppression chamber downcomers during conditions of drywell pressurization. To fulfill this engineered safety feature, proper positioning and operation of the vacuum breakers must be ensured. Therefore, each Pressure Suppression Chamber-Drywell Vacuum Breaker is fitted with redundant pairs of position switches which provide signals of disk position to panel-mounted indicators and redundant annunciators to alarm in the main control room if the disk is open more than the allowable limit.

5.2.3.7 Primary Containment Cooling and Ventilation System

The Primary Containment (drywell) Cooling System utilizes eight fan coil units distributed inside the drywell. See Figure 5.2-18. The Primary Containment Cooling and Ventilation System design parameters are given on Table 5.2-2. Each fan coil unit consists of two cooling coils and two direct-connected motor-driven vaneaxial fans. Each cooling coil is connected to a cooling water supply and return piping system inside the drywell. One or both cooling coils may be utilized for temperature control. Each unit recirculates the drywell atmosphere through the cooling coils to control the drywell space temperature. Cooling water is supplied from the RBCCWS.

Thermocouples are provided to monitor the performance of the drywell cooling system. They are installed in the air and water connections of the drywell coolers as well as the air outlets of the reactor vessel recirculation pump motors as shown on Figure 5.2-18. (The thermocouples on water connections are also shown on Figure 10.5-1). Temperature readouts are provided on indicating panel C-2261A locate outside the drywell.

Fan coil units circulate cooled air around the recirculating pumps and motors, the control rod drive area, and the annular space between the reactor pressure vessel and the biological shield. The personnel access and control rod drive removal openings are sealed to ensure positive flow of cool air from the control rod drive area into the annular space between the reactor vessel and the biological shield, through pipe openings in the reactor vessel support located primarily at the upper level of the control rod drive space.

Cooled air will also be circulated through the reactor vessel head area, the space immediately below the refueling seal plate, and the relief valve area.

Each fan coil unit has provisions for installing dust filters. Filters are to be employed during drywell maintenance activities and will be removed prior to normal station operation.

Cooling water flow to each coil is controlled independently by an electric motorized modulating valve positioned by a valve positioner in the control room. The cooling coil leaving air temperature can be adjusted by regulating the flow of cooling water. A cooling coil failure can be detected by a flow device located in the cooling unit condensate drain line, which is annunciated in the main control room. The standby coil can be put in service and the other isolated by their motorized valves and a check valve in the return line.

Each fan is started from a local panel by using run-off-auto type switches. One fan is started by switching to RUN and the other fan switch is placed in the AUTO position. If the normal operating fan fails, a flow switch will sense a reduced pressure and automatically start the standby fan, light an amber light at a local panel, and annunciate in the control room. Cooling unit discharge air temperature is sensed by a temperature element and indicated in the control room. Upon scram, standby fans will be placed in service automatically to provide additional cooling. All fan coil units can be operated from the emergency power supplies.

The drywell purge ventilation supply system consists of two full capacity fans to supply clean Reactor Building air to the drywell for purge and ventilation purposes, during the reactor shutdown and refueling periods for personnel access and occupancy. The purge exhaust air is normally discharged to the atmosphere through the

Reactor Building exhaust vent. If necessary, SGTS is used for cleanup and the drywell air is exhausted through the main stack.

The ventilation lines supplying the primary containment are provided with two fast acting, pneumatic, cylinder-operated butterfly valves in series for isolation purposes. These valves are normally closed during station operation.

Procedures for normal primary containment venting and purging are established such that gaseous effluent releases from the station remain within the normal release limits. As noted above, purge or vent exhausts can be directed to elevated release points through the Reactor Building vent, or through the SGTS to the main stack.

Drywell and torus purging will normally be conducted to facilitate personnel access subsequent to periods of operation with the primary containment inerted. Primary containment purge operations would normally release on the order of 1 million standard ft^3 of gas. Drywell and torus venting is required during reactor startups in order to maintain normal operational primary containment pressure control as heat loads increase drywell atmosphere temperatures. The volume of gas released during venting operations is expected to be small with respect to purge volumes.

Before purging or venting the containment, airborne contamination levels will be determined and estimates made of expected gaseous activity releases. Selection of release routes and release rates will be made so as to assure compliance with the Technical Specifications. No special area controls or monitoring procedures are imposed during primary containment purging or venting operations.

5.2.3.8 Primary Containment Atmospheric Control System

The Primary Containment Atmospheric Control System (PCACS) has been provided in the design to introduce makeup air or nitrogen into the primary containment.

The capability to operate the primary containment with an inert atmosphere has been provided in the design in accordance with previous licensing commitments. This system is capable of reducing and maintaining the oxygen content of the atmosphere and complies with the requirements set forth by the American Gas Association. The PCACS will be isolated from the primary containment in the event of an accident.

Basically, the equipment in the PCACS performs two functions: (1) initial purging of the primary containment, and (2) providing an automatic supply of makeup gas. If the inerting system is used, the purging equipment converts liquid nitrogen into gaseous nitrogen. Gaseous nitrogen can be introduced into the suppression chamber or the drywell. The PCACS is also capable of automatically providing makeup gas to the primary containment.

5.2.3.9 Drywell Temperature and Pressure Indication

Drywell temperature and pressure are recorded in the main control room. These instruments can be utilized to monitor the essential drywell parameters that are used in the Station Safety Analysis in Section 14.

5.2.3.10 Pressure Suppression Pool Temperature and Level Indication

Pressure suppression pool local and bulk temperature is indicated, recorded and alarmed in the main control room. Pressure suppression pool level is continuously indicated in the main control room. Pressure suppression pool temperature and level can be monitored locally at the Alternate Shutdown Panel C165. These instruments can be utilized to monitor the essential pressure suppression pool parameters that are assumed for initial values in the Station Safety Analysis in Section 14.

5.2.3.11 Drywell Level and Torus Pressure

The Drywell Level and Torus Bottom Pressure are indicated in the Main Control Room. The level indicator measures from plant elevation 47 feet to the containment purge and vent line, elevation 77 feet. The pressure indicator measures from 0 to 100 psig. These parameters are also provided to the EPIC computer. The parameters are used in conjunction with the Emergency Operating Procedures.

5.2.4 Safety Evaluation

5.2.4.1 General

The primary containment and its associated safeguard systems accomplish the following safety design bases:

1. Accommodate the transient pressures and temperatures associated with the postulated equipment failures within the containment (safety design basis 1)

2. Provide a margin for the effects of a metal water and other chemical reactions subsequent to postulated accidents involving loss of coolant (safety design basis 2)

3. Provide a high integrity barrier against leakage of any fission products associated with these equipment failures (safety design basis 3 and 5)

4. Provide for long term core flooding (safety design basis 4)

5. Provide for rapid actuation of the containment barrier (safety design basis 6)

6. Store water for the CSCS (safety design basis 7)

7. Maintain the containment parameters during planned operation to within those assumed in the Station Safety Analysis (safety design basis 8)

These factors are considered in the following evaluation of the integrated PCS.

5.2.4.2 Primary Containment Characteristics Following a Design Basis Accident

In order to establish a design basis for the pressure suppression containment with regard to pressure and temperature rating and steam condensing capability, the maximum rupture size of the Reactor Primary System must be defined. For this design, an instantaneous, circumferential rupture with double ended flow of one recirculation line has been selected as a basis for determining the maximum gross drywell pressure, and the condensing capability of the pressure suppression system. The selection of a failure of this size for the design basis is entirely arbitrary, since the circumferential failure of a recirculation pipe of this magnitude is considered to be of

Although it has been concluded that with the application of conservative piping design and proven engineering practices pipes will not break in such a manner as to bring about movement of pipes sufficient to damage the primary containment vessel, the design of the containment and piping systems does consider the possibility of missiles being generated from the failure of flanged joints such as valve bonnets, valve stems, recirculation pumps, and from instrumentation such as thermowells.

The most positive manner to achieve missile protection is through basic station arrangement such that, if failure should occur, the direction of flight of the missile is away from the containment vessel. The arrangement of station components takes this possibility into account even though such missiles may not have enough energy to penetrate the containment.

Spatial separation and utilization of the biological shield to the maximum extent practical are the measures taken to minimize the possibility of a single potential missile causing a loss of more than one redundant subsection of a vital safety system or a loss of more than one functionally independent safety system.

In order to minimize post-accident containment leakage, the containment penetrations are designed to retain their integrity during postulated accidents.

It is concluded that safety design basis 3 is met.

5.2.4.6 Containment Isolation

One of the basic purposes of the PCS is to provide a minimum of one protective barrier between the reactor core and the environmental surroundings subsequent to an accident involving failure of the piping components of the Reactor Primary System. To fulfill its role as a barrier, the primary containment is designed to remain intact before, during, and subsequent to any LOCA in a process system either inside or outside the primary containment. The process system and the primary containment are considered as separate systems, but where process lines penetrate the containment, the penetration design achieves the same integrity as the primary containment structure itself. The process line isolation valves are designed to achieve the containment function inside the process lines when required.

Since a rupture of a large line penetrating the containment and connecting to the Reactor Coolant System may be postulated to take place at the containment boundary, the isolation valve for that line is required to be located within the containment. This inboard valve in each line is required to be closed automatically on various indications of reactor coolant loss. A certain degree of additional reliability is added if a second valve, located outboard of the containment and as close as practical to it, is included. This second valve also closes automatically on an isolation signal. Both valves shall receive the isolation (closure) signal even if normally closed during reactor operation. If a failure involves one valve, the second valve is available to function as the containment barrier.

By physically separating the two valves there is less likelihood that a failure of one valve would cause a failure of the second. The two valves in series are provided with independent power sources.

The ability of the steam line penetration and the associated steam line isolation valves to fulfill the containment objectives under several postulated break locations in the steam line is described below, and demonstrates the adequacy of the isolation valve design:

1. The failure occurs within the drywell upstream of the inner isolation valve

   Steam from the reactor is released into the drywell and the resulting sequence is similar to that of a design basis LOCA except that the pressure transient is less severe since the blowdown rate is slower. Both isolation valves close upon receipt of the signal indicating low water level in the reactor vessel. This action provides two barriers within the steam pipe passing through the penetration and prevents further flow of steam to the turbine. Thus when the two isolation valves close subsequent to this postulated failure, the primary containment barrier is established, and the reactor is effectively isolated from the external environment

2. The failure occurs within the drywell and renders the inner isolation valve inoperable

   Again the reactor steam will blow down into the primary containment. The outer isolation valve will close upon receipt of the low water level signal, establishing the primary containment barrier

3. The failure occurs downstream of the inner isolation valve either within the drywell or within the guard pipe

   Both isolation valves will close upon receipt of a signal indicating low water level in the reactor vessel. The guard pipe is designed to accommodate such a failure without damage to the drywell penetration bellows, and the design of the pipeline supports protect its welded juncture to the drywell vessel. Thus the reactor vessel is isolated by the closure of the inner isolation valve and the primary containment barrier is established by closure of the outer isolation valve. It should be noted that this condition provides two barriers between the reactor core and the external environment

4. The failure occurs outside the primary containment between the guard pipe and the outer isolation valve

The steam will blow directly into the pipe tunnel through the blowout panel and into the Turbine Building until the isolation valves are automatically closed. Closure of the inner isolation valve places a barrier between the reactor core and the external environment. This barrier serves to isolate the reactor and complete the containment integrity. Closure of the outer isolation valve in this incident serves no useful purpose

5. The failure occurs outside the primary containment and renders the outer isolation valve inoperative

   The primary containment barrier and isolation of the reactor is achieved by closure of the inner isolation valve

6. The failure occurs outside the primary containment between the outer isolation valve and the turbine

   The steam will blow down directly into the pipe tunnel or the Turbine Building until both isolation valves are automatically closed. This action isolates the reactor, establishes the primary containment barrier, and places two barriers in series between the reactor core and the outside environment

The exceptions to the arrangement of isolation valves described above for lines connecting directly to the containment or Reactor Primary System are made only in cases where it leads to a less desirable situation because of required operation or maintenance of the system in which the valves are located. In the cases where, for example, the two isolation valves are located outside the containment, special attention is given to assure that the piping to the isolation valves has an integrity at least equal to the containment.

The TIP system isolation valves are normally closed. When the TIP system cable is inserted, the valve of the selected tube opens automatically and the chamber and cable are inserted. Insertion, calibration, and retraction of the chamber and cable requires approximately 5 min. Retraction requires approximately 1 1/2 min. If closure of the valve is required during calibration, the isolation signal causes the cable to be retracted and the valve to close automatically on completion of cable withdrawal.

It is neither necessary nor desirable that every isolation valve close simultaneously with a common isolation signal. For example, if a process pipe were to rupture in the drywell, it would be important to close all lines which are open to the drywell, and some effluent process lines such as the main steam lines. However, under these conditions, it is essential that containment and core cooling systems be operable. For this reason, specific signals are utilized for isolation of the various process and safeguard systems.

Isolation valves must be closed before significant amounts of fission products are released from the reactor core under DBA conditions.

Because the amount of radioactive materials in the reactor coolant is small, a sufficient limitation of fission product release will be accomplished if the isolation valves are closed before the coolant drops below the top of the core.

It is concluded that safety design basis 6 is met.

5.2.4.7 Containment Flooding

As is discussed in Section 12, the PCS is designed for the conditions associated with flooding the containment.

It is concluded that safety design basis 4 is met.

5.2.4.8 Pressure Suppression Pool Water Storage

Based upon the Station Safety Analysis presented in Section 14 the quantity of water stored in the suppression pool is sufficient to condense the steam from a DBA and to provide water for the CSCS. As discussed in Section 12, the suppression pool is considered in the loading conditions on the PCS.

It is concluded that safety design basis 7 is met.

5.2.4.9 Limitations During Planned Operations

As is discussed in Sections 5.2.3.6, 5.2.3.7, and 5.2.3.8, the PCS is designed to be kept within the limits of parameters assumed in the Station Safety Analysis presented in Section 14 during planned operations.

It is concluded that safety design basis 8 is met.

5.2.4.10 Primary Containment Steam Quenching

The suppression chamber, or torus, is designed to contain a pool of water in order to suppress the pressure during a postulated LOCA by condensing the steam released from the Reactor Primary System. The reactor system energy released by relief valve operation during operating transients also is released into the suppression pool.

As a result of concerns regarding potential instability of steam condensation in a hot suppression pool, the United States Nuclear Regulatory Commission (NRC) has imposed pool temperature limits for plant transients involving safety/relief valve (SRV) operation (Reference 1). The limits which ensure smooth steam condensation for discharge through quenchers are:

1. For all plant transients involving SRV operation during which the steam flux through the quencher perforations exceeds 94 lbm/ft^2-sec, the suppression pool local temperature shall not exceed 200°F.

2. For all plant transients during which the steam flux through the quencher perforations is less than 42 lbm/ft^2-sec, the suppression pool local temperature should be at least 20°F subcooled. This corresponds to a local temperature limit of 201.4°F for PNPS.

3. For plant transients involving SRV operation during which the steam flux through the quencher perforations exceeds 42 lbm/ft^2-sec but is less than 94 lbm/ft^2-sec, the suppression pool local temperature can be established by linearly interpolating the local temperatures established under items (1) and (2) above.

These limits are depicted in Figure 5.2-19.

An analysis was done (Reference 2) to show that PNPS complies with the NRC criteria.

Seven transient events have been identified, one of which is expected to result in the maximum long-term suppression pool temperature. The seven events are as follows:

1A. Stuck-open SRV during power operation with one RHR loop available.

1B. Stuck-open SRV during power operation assuming reactor isolation due to MSIV closure.

2A. Isolation/scram and manual depressurization with one RHR loop available.

2B. Isolation/scram and manual depressurization with the failure of an SRV to reclose (SORV).

2C. Isolation/scram and manual depressurization with two RHR loops available. This case demonstrates the pool temperature responses when an isolation/scram event occurs under normal power operation (i.e., when all systems are operating in normal mode).

3A. Small-break accident (SBA) with manual depressurization; accident mode with one RHR loop available.

3B. Small-break accident (SBA) with manual depressurization and failure of the shutdown cooling system.

The analysis indicated that the maximum temperature occurs during Case 2A. The maximum local pool temperature for this case is 199°F (reference 4) which is less than the 201.4°F limit applicable for low steam flux conditions.

5.2.4.11 Steam Bypass

Following a reactor coolant pipe break inside the containment, the potential exists for the air-steam mixture within the drywell to pass through various leakage paths into the suppression chamber, thereby causing pressurization of the suppression chamber. This increased back pressure in the suppression chamber might lead to an increase in pressurization of the drywell, and possible overpressurization of the containment beyond the design limits.

The bypass area is expressed in terms of $A/\sqrt{k}$, where A is the total bypass (leakage) area and k is the pressure loss coefficient. The maximum allowable leakage area that could exist between the drywell and the suppression chamber is a function of the area of the break as well as the duration of pressurization. The former depends on the P between the drywell and suppression pool, and the latter relates to the time delay until containment sprays are initiated.

In order to assess this relationship, an analysis was performed with various steam break sizes. For large breaks the ΔP is high, but has a short duration. The maximum ΔP results from the DBA. Primary system breaks greater than approximately 0.3 ft^2 will result in rapid depressurization of the primary system. Figure 5.2-22[3] shows the allowable bypass capacity ($A/\sqrt{k}$) as a function of primary system break area.

The allowable $A/\sqrt{k}$ is determined on the basis of the allowable steam mass that can be bypassed without exceeding the containment design pressure of 62 psig. For the Pilgrim Nuclear Power Station the maximum allowable bypass capacity is an $A/\sqrt{k}$ =0.13 ft^2. Typically, the geometric loss factor would be 3 or greater. Thus, the actual allowable bypass area would be approximately 0.2 ft^2. This is equivalent to a 6 in orifice.

When calculating the allowable leakage capacities shown on Figure 5.2-22, the following sequence of events is assumed. Immediately after a break in the primary system, a rapid rise in containment pressure would occur as the noncondensible gases in the drywell are transferred to the suppression chamber. For the allowable leakage calculations, no operator action is assumed until the suppression chamber pressure reaches 35 psig. Further, a 10 min delay is assumed before any action is taken to terminate the transient. In addition to the 10 min operator delay, a 5 min delay is assumed for corrective action to become effective.

The following assumptions were made in calculating the allowable leakage capacities:

1. Flow through the postulated leakage is pure steam. This is a conservative assumption as the amount of steam released into the suppression pool is maximized

2. There is no condensing of the leakage flow on either the suppression pool surface or the torus and vent system structure. This assumption results in a conservative peak pressure calculation

Station emergency procedures ensure that operator corrective action appropriate to the postulated events is taken. If the low-low water level point has not been reached, the operators can depressurize the reactor vessel through the main steam lines to the main condenser or alternately, utilize the relief valves to rapidly reduce reactor pressure. Existing emergency procedures require the initiation of the pressure suppression pool spray mode of the RHRS after verification that the reactor vessel water level is adequate. Further, the procedures require the initiation of the drywell spray mode of the RHRS if the drywell pressure rises to 10 psig.

5.2.5 Inspection and Testing

The following discussion details the surveillance and testing that will be conducted on the various systems or components of the primary containment during construction or station operation.

5.2.5.1 Primary Containment Integrity and Leaktightness

Fabrication procedures, nondestructive testing, and sample coupon tests were made in accordance with the ASME Code of Boilers and Pressure Vessels, Section III, Subsection B. The integrity of the Primary Containment System was verified during construction. The verification included a pneumatic test of the drywell and suppression chamber at 1.25 times their design pressure in accordance with code requirements.

After complete installation of all penetrations in the drywell and suppression chamber, the vessel was pressurized to the calculated peak accident pressure, and measurements taken to verify that the integrated leakage rate from the vessel did not exceed the maximum allowable leak rate. A second test was run at reduced pressure to establish a relationship between leakage rate and containment pressure. The necessary instrumentation is installed in the station to provide the data required to calculate and verify the leakage rate.

Provisions are made so that integrated, containment leakage rate tests may be periodically performed during periods of reactor shutdown, in compliance with 10CFR50, Appendix J, Primary Containment Leakage Testing for Water Cooled Power Reactors.

5.2.5.2 Penetrations

The design permits the testing of penetrations which have resilient seals or expansion bellows without pressurizing the entire containment system. Leak detection may then be accomplished either by the use of soap suds, pressure decay techniques, or other acceptable methods.

Pipe penetrations which must accommodate thermal movement are provided with two ply bellows expansion joints. These two ply bellows are provided with test taps so that the space between the plies can be pressurized to the calculated peak accident pressure to permit testing of the individual penetrations for leakage.

Electrical penetrations are also separately testable. The test taps are located so that the tests of the electrical penetrations can be conducted without entering or pressurizing the drywell or suppression chamber.

All containment closures which are fitted with resilient seals or gaskets are separately testable to verify leaktightness. The covers on flanged closures, such as the equipment access hatches, the drywell head, access manholes, and personnel air lock doors, are provided with double seals, and with a test tap which allows pressurization of the space between the seals without pressurizing the entire containment system.

5.2.5.3 Isolation Valves

The test capabilities which are incorporated in the PCS to permit leak detection testing of containment isolation valves are separated into two categories.

The first category consists of those pipelines which open into the containment atmosphere and do not terminate in closed loops outside the containment, and contain two isolation valves in series. Test taps are provided between the two valves which permit leakage monitoring of the first valve when the containment is pressurized.

The test tap can also be used to pressurize between the two valves to permit leakage testing of both valves simultaneously.

The second category consists of those pipelines which connect to the Reactor System and contain two isolation valves in series. A leakoff line is provided between the two valves, and a drain line is provided downstream of the outboard valve. This arrangement permits monitoring of leakage on the inboard and outboard valves during Reactor System hydrostatic tests, which can be conducted at pressures up to the reactor system operating pressure of 1,000 psig.

Generally, leakage testing is not required for isolation valves contained in pipelines whose terminal end will remain submerged in the suppression pool throughout the duration of the design basis LOCA. Therefore, these valves are not relied upon to prevent the release of fission products, and therefore do not perform a containment isolation function. Reference Number 5, Section 5.2.9 provides a list of valves which fit the above category.

Isolation valve closing time is determined during the functional performance test performed prior to reactor startup.

5.2.6 Nuclear Safety Requirements for Plant Operation

The entries in this section represent the nuclear safety requirements for the PCS for each BWR operating state which represents an extension of the stationwide BWR systems analysis of Appendix G. The following referenced portions of the safety analysis report provide information justifying the entries in this section:

1	Reference	Information Provided
1.	Preceding portions of Section 5.2	Description of PCS
2.	Section 7.2, Primary Containment and Reactor Vessel Isolation Control System	Description of PCICS
3.	Station Safety Analysis, Section 14	Analyses verifying primary containment responses and radiological effects of postulated accidents
4.	Station Nuclear Safety Operational Analysis, Appendix G	Identification of conditions and events for which PCS is required
5.	Bodega Bay Preliminary Hazards Summary Report, Appendix 1, Docket 50-205	Pressure suppression test information

6. Jacobs, I.M., Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards. General Electric Co., Atomic Power Equipment Department, APED-5736, April 1969	Describes methods used to establish allowable repair times

Each detailed requirement in the following analysis is referenced, if possible, to the most significant station condition originating a need for the requirements by identifying a matrix block on one of the Matrix 3 sheets of Table G.5-3. The matrix block referenced is given in parentheses beneath the detailed requirements in the "minimum required for action" section.

The matrix block references identify the BWR operating state, the event number, and the system number. For example, F39-82, identifies BWR operation state F (Matrix 3), event (row) No. 39, and system (column) No. 82.

Minimum Required for Action

1. Vacuum Relief System
   (C39-109) (E39-109)
   (D39-109) (F39-109)

2. Primary Containment
   (C39-82) (E39-82)
   (D39-82) (F39-82)

3. Drywell Pressure and Temperature Indicators
   (C2-103) (E2-103)
   (D3-103) (F4-103)

4. Pressure Suppression Pool Water Level and Temperature Indicators
   (A35-104) (D3-104) (F4-104)
   (B35-104) (D39-104) (F39-104)
   (C2-104) (E2-104)
   (C39-104) (E39-104)

5. Pressure Suppression Pool Water Storage
   (A35-83) (D39-83)
   (B35-83) (E39-83)
   (C39-83) (F39-83)

Requirements are placed on the operating status of systems essential to containment to assure their availability to control the release of any radioactive material from irradiated fuel in the event of an accident condition. The PCS provides a barrier against uncontrolled release of fission products to the environs in the event of a break in the Reactor Coolant Systems. Whenever the reactor is in states C, D, E, and F (with nuclear system pressurized), failure of the Reactor Coolant System could cause rapid expulsion of the coolant from the reactor, with an associated pressure rise in the primary containment. Primary containment is required, therefore, to limit the release of fission products to the station environs so that offsite doses would be well below the values specified in 10CFR100.

The calculated radiological doses given in Section 14 were based on an assumed leakage rate of 0.5 percent. Increasing the assumed leakage rate at 56 psig to 2.0 percent would increase those doses by approximately a factor of four, still leaving a substantial margin between the calculated dose and the 10CFR100 regulation.

The suppression pool water volume provides the heat sink for the Reactor Coolant System energy released following the LOCA. In states A and B the suppression pool water is available as a source of makeup water to replace possible leakage from the reactor vessel and primary system.

The maximum water volume limit allows for an operating range without significantly affecting the accident analyses with respect to free air volume in the suppression pool. The maximum pool bulk temperature of 130°F would accommodate a complete accident blowdown with minimum water volume without exceeding the design temperature limit of 170°F immediately after blowdown. The design minimum water temperature of 40°F assures that the water is always in the liquid state. Suppression pool temperature limits have been selected to assure reactor depressurization can occur without high pressure suppression chamber loadings caused by instability during steam condensation.

The Drywell Suppression Pool Vacuum Breaker System is required to prevent water oscillation in the downcomers due to low steam flow rates in the downcomers, and to provide protection against negative pressure conditions in the containment vessel. Allowing one valve to be inoperative reduces the total vacuum relief area by only 10 percent. Since the valves are totally enclosed within the containment, possible leakage through them does not affect the containment system leakage.

The Suppression Pool Reactor Building Vacuum Relief System assures that the primary containment is not operated at a significant negative pressure relative to its surroundings. The 0.5 psi differential pressure setting was chosen on the basis of Relief System pressure drop, valve opening times, and peak mass flow to limit the external pressure on the suppression chamber to less than its design value of

2.0 psig. The Vacuum Relief System is a redundant system and full relief capacity is available through either valve. If one vacuum breaker or its block valve becomes inoperable, there is no immediate threat to primary containment integrity, thus, reactor operation may continue while repairs are being made, provided the repair procedure does not violate primary containment integrity. Possible leakage of these valves is included in the containment system integrated leakage rate tests performed periodically.

5.2.7 Current Technical Specifications

The current limiting conditions for operation, surveillance requirements, and their bases are contained in the Technical Specifications referenced in Appendix B.

5.2.8 Pipe Break Transient Analysis

5.2.8.1 Pipe Mechanical Failure and Safety Design

The Pilgrim Nuclear Power Station primary containment satisfies safety design basis 1 by its capability "to accommodate the transient pressures and temperatures associated with the postulated equipment failures within the containment." The intent of safety design basis 1 is to provide a basis for determining the primary containment internal design pressure and associated temperature, and that basis is that the primary containment must remain functional after accommodating the largest mass flow and energy release associated with the design basis LOCA. See Section 5.2.4.1.

Section 5.2.4.2 discusses the selection of the failure conditions that establish containment design parameters. The capability to satisfy safety design basis 1 is demonstrated in the primary containment response analysis to the design basis LOCA present in the Station Safety Analysis in Section 14. It is not the intent, nor has it ever been assumed that it would be the intent, that safety design basis 1 be used as a basis for evaluating the abilities of the primary containment to accommodate the mechanical forces and energies that might be associated with the movement of unrestrained pipe during a postulated LOCA.

INTENTIONALLY LEFT BLANK

INTENTIONALLY LEFT BLANK

Section 5.2.4.5 discusses containment integrity protection within the scope and the intent of safety design basis 3 and defines the pertinent loading considerations that have been evaluated in order to meet safety design basis 3.

In order to minimize the probability of an instantaneous failure in the Reactor Primary System piping, design provisions were made to minimize or identify conditions that could lead to such a failure.

As discussed in Appendices J.2.4 and F.2.6, the Reactor Primary System is designed to meet the intent of Criterion 35 of the Proposed AEC General Design Criteria, thus reducing even further the extremely low probability of an instantaneous piping failure due to brittle fracture.

In addition, a Nuclear System Leak Detection System, as described in Section 4.10, is provided to identify primary system leakage rates well below those leakage rates which correspond to the critical size for rapid crack propagation. The capability of this system to identify these leakage rates will provide significant protection against an instantaneous primary system piping failure due to crack propagation by allowing station personnel sufficient time to take appropriate corrective measures. Supplementary protection will be provided by the comprehensive inservice inspection program discussed in Appendix K.

In conclusion, the design fabrication, testing, and inspection of the Reactor Primary System has emphasized the elimination of potential causes for instantaneous piping failures, and thus obviates the need to design the PCS to withstand the mechanical effects of the failed pipe. Therefore, the protection of the primary containment from the mechanical effects of an unrestrained failed pipe is not a safety design basis for Pilgrim Nuclear Power Station.

5.2.8.2 Pipe Protection System

As a result of an investigation, selected areas of the interior of the drywell shell were protected to reduce the possibility of breaching of the primary containment by postulated failure of a large, unrestrained pipe in the primary pressure boundary.

All pipe penetrations through the drywell have been designed to withstand the forces and moments resulting from a pipe rupture inside the drywell. Main steam and feedwater piping have restraints outside the drywell for protection of the penetration assemblies and outboard isolation valves.

The piping systems considered for postulated failure and having the potential to breach the containment are those which are located within the spherical section of the drywell and normally pressurized to reactor pressure (main steam, HPCI steam supply, feedwater, RHR). These large pipes are postulated to fail at circumferential butt welds with the jet reaction force acting normal to the rupture surface, and resulting in pipe rotation around a plastic hinge. The drywell areas requiring protection are shown on Figure 5.2-23, and are those areas in the spherical section where the single postulated large pipe weld failure could result in pipe movement to the extent that the ruptured pipe could contact the interior of the drywell shell with sufficient energy to perforate the drywell.

The protection system consists generally of steel members attached to a reinforcing plate. The protection system is arranged to receive a postulated rupture pipe, absorb a portion of the impact energy, and distribute the impact load over an area of the drywell shell such that the combined energy absorbtion capacity of the protection system and the drywell shell is greater than the impact energy of the ruptured pipe. The protection system and the drywell shell will deform through the 2 in air gap between the drywell and the concrete shield without causing breaching of the drywell. Details of the protection system are shown on Figure 5.2-23.

Areas of the spherical section of the drywell shell requiring protection have been determined by plotting the potential area at which each ruptured pipe end could contact the drywell when the pipe is rotated around various possible plastic hinge points. The force causing pipe movement and deformation around a plastic hinge is the jet reaction resulting from blowdown from the reactor system.

The impact energy of a ruptured pipe has been determined as a function of the jet reaction force, pipe plastic bending moment, and the configuration of the pipe with respect to the drywell shell. The energy required for perforation of the drywell shell has been determined from an empirical relation developed from a series of experiments using steel projectiles.

The protection system component size and placement is based upon the requirement to distribute the pipe impact energy over a sufficient area of the drywell shell, such that the combined energy absorbing capability of the protection system and drywell shell is greater than the impact energy of the ruptured pipe. The steel beams are arranged to minimize the possibility of a ruptured pipe end from resulting in a localized load bearing directly on the drywell shell. The beams are attached to a steel plate located between the beams and the drywell shell. This plate results in increasing the energy absorbing capability of the drywell by increasing the impact area, and increasing the effective thickness of the drywell shell. The plate also serves as a means of restraining the beams against potential jet impingement loads and the component of pipe impact loads tangential to the drywell shell.

The protection system is attached to the drywell shell at the weld pads with additional support from the floor structures as required. The protection system supports are designed to withstand the loads from the Safe Shutdown Earthquake.

The protection system components have been selected and located such that maximum protection is provided for the drywell shell against postulated pipe ruptures with minimum interference to required access for inservice inspection.

Pipe ruptures within the cylindrical section of the drywell have been considered and no protection is required because:

1. Pipe movement distances to contact the drywell are insufficient to obtain an impact energy exceeding the energy required to perforate the 1 1/4 in shell thickness

2. The close proximity of the drywell shell to the piping systems is such that pipe rotation around a plastic hinge is insufficient to result in the ruptured end becoming a localized load on the drywell

The analysis and basis for design of the protection system is conservative because:

1. Jet reaction forces have not been reduced due to the throttling effect of partial pipe closure at the plastic hinge point

2. Pipe impact energies have not been reduced by the energy absorbed by pipe deformation at the point of contact between the protection system and the drywell shell

3. Impact energy required to perforate the drywell shell is based on test data using tool steel projectiles, and is therefore lower than the energies required for perforation with typical pipe materials

5.2.8.3 Design Basis Line Break

The design basis steam line break accident (SLBA) is described in detail in Section 14. This accident is assumed to result in a complete guillotine break of the main steam line, resulting in a 1.74 ft^2 break area. All other breaks in piping attached to the vessel above the core result in peak clad temperatures which are lower than those resulting from the SLBA. Since there are no perforations for the SLBA, there will be none for smaller steam line breaks. While the SLBA evaluated in Section 14 considers isolation of the reactor vessel, an analysis of an SLBA inside the primary containment (i.e., no isolation) is described in Section 5.2.3.2. The results of this analysis show that the core will remain covered throughout the transient and the resultant peak clad temperature will be less than normal operating temperatures, which are well below the temperature where clad perforation could occur. As in the case of the design basis SLBA, all other smaller steam lines which could fail in such a manner that isolation is not achieved would also not result in clad perforations. Consideration of those liquid breaks which could conceivably result in containment breaching as a result of pipe whip has also resulted in the conclusion that no fuel perforations will occur. In particular, for the feedwater line break (approximately 0.5 ft^2) incore is also not uncovered. It can therefore be concluded that the resultant radiological exposures for the above pipe failures will at the maximum be based on only that activity contained in the primary coolant, which is discharged to the secondary containment.

To provide an upper limit to the radiological exposures, the assumptions have been made that:

1. All of the primary coolant which contained activity is eventually discharged to the secondary containment

2. Considering the thermodynamics of the coolant discharged, a maximum of 1/3 of the coolant is flashed to steam resulting in the release to the secondary containment of 1/3 of the coolant activity

3. Consideration of the condensing and plateout surfaces that the released steam will have to come in contact with prior to being released from the top of the Reactor Building results in a minimum reduction factor of 3 for the released iodine activity

4. The activity is released from the top of the Reactor Building under those meteorological conditions, which maximize the offsite exposures

5. The activity contained in the reactor coolant is consistent with an offgas emission rate of 10^5 microcuries/sec

Based on the above considerations, the resultant site boundary thyroid dose is 0.08 rem while the LPZ thyroid dose is 0.002 rem. If the conservative assumption is made that downwash of the released effluent occurs and that the coolant activity is at a level consistent with the technical specification offgas activity (i.e., 0.9 ci/sec), the resultant site boundary thyroid dose is 15 rem and the LPZ thyroid dose is 0.6 rem, both of which are well below the 300 rem guideline set forth in 10CFR100.

5.2.9 References

1. Bodega Bay Preliminary Hazards Report, Appendix I, Docket 50-205, December 28, 1962.

2. General Electric Company, "PNPS Unit 1 Suppression Pool Temperature Response," NEDC-22089-P, March 1982.

3. J. M. Caroll, BECo Letters to NRC, May 15, 1973.

4. General Electric Company, "Pilgrim Suppression Pool Temperature Analysis," Letters from R. Thibault to G. McHugh, December 1982.

5. Franklin Research Center Technical Evaluation Report, "Containment Leakage Rate Testing", TER-C5257-40, May 5, 1981.

6. NRC letter, "Licensee Response to IE Bulletin 79-08 and Acceptability of Single Check Valves as Containment Isolation for Pilgrim," Ronald Eaton (NRC) to G. W. Davis, February 4, 1991.

7. E. H. Hoffman. et. al., General Electric Company, "Drywell Temperature Analysis for Pilgrim Nuclear Power Station," EAS-98-0887; August, 1987.

8. Letter, GE to BECo, "Safety Evaluation of Proposed Capping of Certain Drywell Spray Nozzles," G-HK-7-157, dated April 20, 1987.

TABLE 5.2-1

PRIMARY CONTAINMENT SYSTEM
PRINCIPAL DESIGN PARAMETERS AND CHARACTERISTICS

Pressure suppression chamber:	
Internal design pressure	+56 psig
External design pressure	+2 psig
Drywell:	
Internal design pressure	+56 psig
External design pressure	+2 psig
Drywell free volume	(approx) 147,000 ft^3
Pressure suppression chamber free volume	(approx) 120,000 ft^3
Pressure suppression pool water volume, maximum	(approx) 94,000 ft^3
Pressure suppression pool water volume, minimum	(approx) 84,000 ft^3
Submergence of vent pipe below pressure suppression pool surface	(approx) 3.00 to 3.25 ft
Design temperature of drywell	281°F
Design temperature of pressure suppression chamber	281°F
Downcomer vent pressure loss factor	6.21
Break area/total vent area	0.0194
Drywell free volume/pressure suppression chamber free volume	1.34
Primary system volume/pressure suppression pool volume	0.268
Drywell free volume/primary system volume	7.4
Calculated maximum pressure during blowdown:	
Drywell	45 psig
Pressure suppression chamber	27 psig
Initial pressure suppression chamber temperature rise	35°F

TABLE 5.2-2

DRYWELL ATMOSPHERE COOLING DATA SHEET

Location	Average	Maximum*
General	135°F	148°F
Recirculation Pump Motor Area	-	128°F
Entering Air Temperature to Cooling Units	135°F	148°F
Leaving Air Temperature from Cooling Units	85°F	95°F
Cooling Water Supply Temperature	75°F	85°F
Cooling Water Return Temperature	90°F	100°F
Drywell Heat Gain	2.4×10^6 Btu/hr	3.4×10^6 Btu/hr
Total Cooling Unit Capacity	3.6×10^6 Btu/hr	5.6×10^6 Btu/hr
Total Cooling Unit Fan Capacity	72,000 ft^3/min	110,000 ft^3/min
Total Fan Brake hp	54.8	67.8
Drywell Temperature 10 hr after shutdown	105°F	105°F

NOTE:

*As a result of higher cooling water supply temperature and extra heat load from scram of the control rod drives.

5.3 SECONDARY CONTAINMENT SYSTEM

5.3.1 Safety Objective

The safety objective of the Secondary Containment System (SCS), in conjunction with other engineered safeguards and nuclear safety systems, is to limit the release to the environs of radioactive materials so that offsite doses from a postulated DBA will be below the guideline values of 10CFR100.

5.3.2 Safety Design Basis

The safety design bases of the SCS are as follows:

1. The SCS shall be designed to provide secondary containment when the primary containment is operable and when the primary containment is open

2. The SCS is designed with sufficient redundancy so that no single active system component failure can prevent the system from achieving its safety objective

3. The SCS shall be designed in accordance with Class I design criteria. (Exception to this is the containment access locks. Since simultaneous LOCA's and SSE's are not postulated, the access locks shall be designed in accordance with Class II design criteria. The access lock door lying directly in the SCS shall be designed in accordance with Class I design criteria so that the possibility of a ground level release to the environs through the access locks is eliminated if a seismic event were to follow or precede an accident which results in a contaminated reactor building atmosphere.) The SCS is not designed to withstand tornado loads

4. The secondary containment shall be designed to limit the ground level release to the environs of airborne radioactive materials so that offsite doses from a design basis fuel handling, or loss of coolant accident (LOCA) will be below the guideline values stated in 10CFR100

5. The Reactor Building shall be designed to contain a positive internal pressure of at least 7 in of water

6. The SCS shall be designed to be sufficiently leaktight to allow the Standby Gas Treatment System (SGTS) to reduce the Reactor Building pressure to a minimum subatmospheric pressure of 0.25 in of water, under neutral wind conditions, when the SGTS fans are exhausting Reactor Building atmosphere at a maximum of 4,000 ft^3/min

7. The Reactor Building Isolation and Control System (RBICS) shall be designed to isolate the Reactor Building sufficiently fast to prevent fission products from the postulated fuel handling accident from being released to the

environs through the normal discharge path

8. The SCS is provided with means to conduct periodic tests to verify system performance

5.3.3 Description

5.3.3.1 General

The SCS consists of four subsystems. These subsystems are the Reactor Building, the RBICS, the SGTS, and the main stack. The SCS surrounds the Primary Containment System, and is designed to provide secondary containment for the postulated LOCA. The SCS also surrounds the refueling facilities and is designed to provide primary containment for the postulated refueling accident.

The SCS utilizes four different features to mitigate the consequences of a postulated LOCA (pipe break inside the drywell) and the refueling accident (fuel handling accident). The first feature is a negative pressure barrier which minimizes the ground level release of fission products by exfiltration. The second feature is a low leakage containment volume which provides a holdup time for fission product decay prior to release. The third feature is the removal of particulates and iodines by filtration prior to release. The fourth feature is the exhausting of the secondary containment atmosphere through an elevated release point which aids in dispersion of the effluent by atmospheric diffusion. Each of the features is provided by a different combination of subsystems: the first by the Reactor Building, the RBICS, and the SGTS exhaust fans; the second by the Reactor Building and the RBICS; the third by the SGTS filters; and the fourth by the main stack.

5.3.3.2 Reactor Building

The Reactor Building completely encloses the reactor and its pressure suppression Primary Containment System. The Reactor Building houses the refueling and reactor servicing equipment, new and spent fuel storage facilities, and other reactor auxiliary and service equipment. Also housed within the Reactor Building are the CSCS, Reactor Cleanup Demineralizer System, Standby Liquid Control System (SLCS), Control Rod Drive (CRD) System, Reactor Protection System (RPS), and electrical equipment components.

The structural design features of the Reactor Building are described in Section 12. Discussions of the Reactor Building's Class I design are included in Section 12 and Appendix C. The Reactor Building is also designed to meet the shielding requirements discussed in Section 12.

5.3.3.3 Reactor Building Isolation and Control System

The RBICS serves to trip the Reactor Building supply and exhaust fans, isolate the normal ventilation system, and provide the starting signals for the SGTS in the event of the postulated LOCA inside the drywell, or the postulated fuel handling accident in the Reactor Building. Either of two signals will initiate the SCS. These signals, which indicate a LOCA inside the drywell, are high drywell pressure or low reactor water level. In addition, radiation monitors in the operating (refueling) floor ventilation exhaust duct, which indicate a fuel handling accident, can initiate the SCS. Secondary containment can also be initiated manually from the control room.

Normally open, air-operated isolation dampers are provided on the discharge side of the Reactor Building and operating floor supply fans. Normally open, air-operated isolated dampers are provided on the intakes to the operating floor ventilation exhaust fans, the clean area exhaust fans, the contaminated area exhaust fans (upstream of the filter assemblies), and the control rod drive maintenance room exhaust fan. See Figure 5.3-1. Two dampers in series are provided throughout the isolation system to provide the required redundancy. Both dampers fail closed upon loss of dc power to the solenoids or upon loss of instrument air to the dampers. The isolation dampers are piston operated and designed to close after receipt of the secondary containment initiation signal to prevent release of radioactive material from the secondary containment. The refueling floor exhaust isolation dampers must close in 3 sec to isolate the most direct path outside secondary containment. A 5 sec closing time is sufficient for the remaining dampers.

Penetrations of the secondary containment are designed to have leakage characteristics consistent with secondary containment leakage requirements. Electrical penetrations in the Reactor Building are designed to withstand normal environmental conditions and to retain their integrity during the postulated fuel handling accident and the LOCA inside the drywell. Two interlocked sealed doors on the equipment and personnel access locks assure that building access can not interfere with maintaining the secondary containment integrity.

All normally open drains which are open both to the secondary containment and the outside atmosphere are provided with water seals to maintain containment integrity. This is exemplified by the four 14 in dewatering lines for the reactor auxiliary bay floor sumps. These lines penetrate the secondary containment boundary, two below each of the two sumps, and terminate in a pair of troughs within the torus compartment. The two 4 ft cubic shaped troughs, located adjacent to the east wall, maintain containment integrity by providing water seals for each of the four lines. High and low levels are alarmed in the control room. On low level, the operators are directed by procedure to refill the troughs, to ensure containment integrity.

5.3.3.4 Standby Gas Treatment System

The SGTS consists of two, similar, parallel air filtration assemblies separated by an 18 in thick concrete block wall and completely enclosed within a Class I structure. Each of the filtration assemblies are full capacity. Each consists of a demister (use of the demister is optional), an electrical heating coil, a high efficiency particulate absorber (HEPA), two charcoal filter beds, and a final HEPA filter. With the Reactor Building isolated, each of the two fans has the necessary capacity to reduce and hold the building at a minimum subatmospheric pressure of 0.25 in of water.

Each fan has a design flow rate of 4,000 ft^3/min. Motor-operated exhaust fan outlet damper controls are provided to maintain the required negative pressure. See Figure 5.2-17. The system consists of two filter banks. Loss of dc power and/or supply air to the solenoids causes the valves in filter bank A to open and the valves in filter bank B to close. A dedicated air system is provided for long-term damper actuation. The air system consists of two high pressure air tanks, a pressure reducing regulator with inlet and outlet pressure gages, relief valve, two low pressure gages, one low pressure switch, four low pressure air receivers, solenoids valves and manual valves, and tubing required to make a complete system. The low pressure section of the system is designed to Class I design criteria. The high pressure section is designed to Class II design criteria. The low pressure system of the dedicated air system is designed to provide two complete actuation cycles of the SGTS filter isolation dampers/valves and allow for air system leakage for long-term operation.

The demister is designed to remove entrained water droplets and mist from the entering air stream. Use of the demister is optional since there are no design basis accident scenarios which would expose the SGTS to water droplets or mist. The electric heating coil is designed to reduce the relative humidity of the air stream to 70 percent. An interlock with its associated exhaust fan prevents the heating coil from operating when the fan is shut down. Each HEPA filter is designed to be capable of removing at least 99.97 percent of the 0.30 micron particles which impinge on the filter. The charcoal filters are iodide-impregnated activated carbon filters capable of removing in excess of 99 percent of the iodine in the air stream with 10 percent of the iodine in the form of methyl iodide (CH_3I) under entering conditions of 70 percent relative humidity.

The accident evaluations using the standard NRC approach are described in Section 14.9. In these analyses the SGTS charcoal filters were credited with removal of 95 percent of the influent iodine.

The system will start automatically upon a high radiation signal from the operation (refueling) floor ventilation exhaust duct monitor, or upon receipt of high drywell pressure or low reactor water level signals. The system can also be manually started from the control room. Upon receipt of any of the initiation signals. The AUTO "A" train fan and heater start and its associated isolation valves open, the STANDBY "B" train suction valve opens, starting the "B" train fan and heater and in turn, opening the "B" train discharge isolation valve. In the event that the "A" train is down for maintenance the STANDBY "B" train suction valve opens when the train mode switch is in the MAINTENANCE position. Each fan and heater draws air from the isolated reactor building at a flow rate up to approximately 4000 ft^3/min. After a preset time delay period, the "B" train suction valve closes, which in turn trips the "B" train fan and heater, closing its discharge valve. Cross-connections between the filter trains are provided to maintain the required decay heat removal cooling air flow of low humidity air on the charcoal filters in the inactive treatment train.

The STANDBY "B" filter train is automatically started in the event of an AUTO "A" train heater, fan, air supply and/or power failure. In addition, the "B" train is automatically started in the STANDBY mode, whenever the STANDBY "B" train suction valve is not fully CLOSED. The system discharges to the main stack through a 20 in underground line. The SGTS fans are powered from the emergency service portions of the auxiliary power distribution system.

Drywell and torus purge exhaust can also be directed to the SGTS for processing before release up the main stack. See Section 5.2.

The High Pressure Coolant Injection System (HPCI) gland seal steam condenser exhauster discharge is also routed to the SGTS during accident conditions. The Reactor Building Heating and Ventilating System is discussed in Section 10.9.

During a severe accident, the torus can be directly vented to the main stack bypassing the SGTS. See Section 5.4.7.

5.3.3.5 Main Stack

The location of the main stack is shown on Figure 1.6-1. The top of the stack is at elevation 400 ft msl. The structural design of the stack is discussed in Section 12.

5.3.4 Safety Evaluation

The SCS provides the principal mechanisms for the mitigation of the consequences of an accident in th Reactor Building. The primary and secondary containment act together to provide the principal mechanisms for the mitigation of the consequences of an accident in the drywell. If the leakage rate of the building is low, and the leakage air is filtered and discharged to the elevated release point (utilizing the SGTS and the main stack) the offsite radiation doses that result from postulated accidents are reduced significantly. The Reactor Building is a Class I structure (with the exception of the secondary containment access locks which are Class II structures) designed in accordance with all applicable codes. Design of the Reactor Building for a maximum inleakage rate of 4,000 ft^3/min at a building subatmospheric pressure of 0.25 in of water at neutral wind conditions, results in a low exfiltration rate even during high wind conditions.

In the event of a pipe break inside the primary containment or a fuel handling accident, Reactor Building isolation will be effected and the SGTS will be initiated. Both SGTS exhaust fans will start. After a preset time delay, one fan is stopped.

With the Reactor Building isolated, each fan in the SGTS has the capability to hold the building at a subatmospheric pressure of 0.25 in of water when drawing air from the building at a flow rate of 4,000 ft^3/min. Exhaust fan outlet damper controls on each fan are provided to maintain the required flow rate.

The RBICS performs the required isolation actions of the SCS following receipt of the appropriate initiation signals. Following initiation, the Reactor Building ventilation isolation dampers close within a specified time to prevent release of radioactive material from the secondary containment. The refueling floor exhaust isolation dampers must close in 3 sec to isolate the most direct path outside secondary containment. A 5 sec closing time is sufficient for the remaining dampers. The RBICS also automatically trips the Reactor Building supply and exhaust fans, and starts the SGTS. The normal design flow rate in the Reactor Building operating (refueling) floor exhaust duct is 40,000 ft^3/min. During shutdowns, the flow rate is increased to approximately 50,000 ft^3/min at which time it takes more than 3 sec for fission products released in any postulated fuel handling accident to travel from the operating (refueling) floor ventilation exhaust radiation monitors to the isolation dampers. Thus, no direct release of fission products to the environment (bypassing the SGTS filtration processes, and the elevated release point provided by the main stack) is possible, except when the direct torus vent path is used following a beyond design basis accident.

The SGTS filters exhaust air from the Reactor Building and discharges the processed air to the main stack. The system filters particulates and iodines from the air stream in order to reduce the level of airborne contamination released to the environs via the main stack. When the system is exhausting from the Reactor Building, the building is held at a minimum subatmospheric pressure of 0.25 in of water.

Appendix G identifies requirements for establishing secondary containment (Safety Action 27), following an assumed pipe break inside the primary containment (Event 39), and following an assumed spent fuel handling accident (Event 40). Secondary containment is not established following assumed pipe failures which result in the release of steam into the Reactor Building (Event 41).

The following piping which is located within the Reactor Building and normally contains hot fluids at reactor pressure was considered: High Pressure Coolant Injection (HPCI) turbine steam supply line; Reactor Core Isolation Cooling (RCIC) turbine steam supply line; and Reactor Water Cleanup System (RWCU) piping.

The maximum rate of steam release into the Reactor Building and the corresponding period of steam release was calculated for the above piping:

1. HPCI steam line:
   Maximum release rate, 300 lb/sec of steam
   Period of release, 22 sec

2. RCIC steam line:
   Maximum release rate, 25 lb/sec of steam
   Period of release, 17 sec

3. RWCU piping:
   Maximum release rate, 250 lb/sec of steam
   Period of release, 22 sec

Steam leakage into the Reactor Building could be exhausted through the ventilation exhaust systems operating at normal building pressures at a calculated rate of 63 lb/sec of steam. The SGTS operating at normal Reactor Building pressures could exhaust about 5 lb/sec of steam. Steam leakage in excess of these amounts would result in Reactor Building pressure increases above normal.

The Reactor Building is designed to relieve excessive internal pressures so as to preserve main structural integrity, considering the rapid pressure reduction outside the building associated with tornadoes. Refer to Appendix H.5. Reactor Building differential pressures exceeding about 0.5 psi will be relieved through the Reactor Building roof.

Steam leakage within the normal operating capability of the ventilation exhaust systems would be ducted to the main building exhaust vent. The ventilation exhaust from the principal Reactor Building compartments housing the RCIC steam supply line and turbine, the HPCI steam supply line and turbine, and the RWCU are monitored by temperature elements. These elements provide temperature indication and high temperature alarms in the main control room. The temperature set points for these alarms will alert the operator to potential steam leakage conditions at leakage rates that are less than the normal operating capability of the ventilation exhaust systems.

Steam leakage rates that exceed the capability of the normal operating ventilation exhaust systems could result in abnormal ventilation flow paths and abnormal Reactor Building exhaust locations. The design of the Reactor Building would indicate that likely abnormal release locations would include the building roof and building access locations.

Steam leakage into a compartment within the operating capability of the ventilation exhaust systems would be confined within the normal exhaust paths, and therefore would limit the steam flooding principally to the compartment where the leakage originated. Thus the operability of safety related equipment, controls, and instrumentation located in other compartments would be maintained.

The ventilation exhaust temperature sensors will detect steam leakage from the RCIC steam line, the HPCI steam line, or the RWCU piping at leakage rates that are below the normal operating capability of the ventilation exhaust from the compartments housing these hot, pressurized lines. Early detection of steam leakage at rates below the capability of the normal ventilation systems and subsequent isolation of leaks provide protection of safety related equipment within the Reactor Building. See Section 7.3.

The main stack provides an elevated release point for airborne activity during the postulated station loss of coolant and refueling accidents. Release of activity to the environs from the Secondary Containment System is analyzed in detail in Section 14, Station Safety Analysis. It is concluded that the safety design bases are met.

5.3.5 Inspection and Testing

The secondary containment leakage rate can be determined in the following manner. The Reactor Building is isolated and the SGTS is started with one treatment train and its associated exhaust fan. The exhaust flow rate is controlled by the fan outlet damper control position as determined by flow rate measurements in the SGTS exhaust duct. The fan outlet damper positioner is used to control the exhaust flow rate at 4,000 ft^3/min.

If the subatmospheric pressure as measured within the Reactor Building is equal to or exceeds 0.25 in of water (with neutral wind conditions at the site) the building safety design basis leaktightness with respect to inleakage is verified.

Tests of the ability of the various isolation initiation signals to automatically render the Reactor Building isolated, to trip the supply and exhaust fans, and to start the SGTS can be conducted by simulating the isolation signals.

Provisions are made for periodic tests of each filter unit. These tests include determinations of differential pressure across each filter and of filter efficiency. Connections for testing, such as injection and sampling, are located to provide adequate mixing of the injected fluid and representative sampling and monitoring, so that test results are indicative of performance. Each HEPA is tested with DOP (Di-octyl-phthalate) smoke. The charcoal filters can be tested for bypass with freon.

The electric heating coil in each filter train is tested to show that the relative humidity of an entering air stream is reduced.

5.3.6 Nuclear Safety Requirements for Plant Operation

General

The entries in this section represent the nuclear safety requirements for the SCS for each BWR operating state which represents an extension of the stationwide BWR systems analysis of Appendix G. The following referenced portions of the Safety Analysis Report provide important information justifying the entries in this section:

Reference	Information Provided
1. Earlier parts of Section 5.3	Description of the SCS
2. Station Nuclear Safety Operational Analysis Appendix G	Identifies conditions and events for which the SCS is required
3. Jacobs, I.M. Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards. General Electric Co., Atomic Power Equipment Department, APED - 5736, April 1969	Describes methods used to establish allowable repair times

Each detailed requirement in this section is referenced, if possible, to the most significant station condition originating the need for the requirement by identifying a matrix block on one of the Matrix 3 sheets of Table G.5-3. The matrix block references are given in parentheses beneath the detailed requirements in the "minimum required for action" section.

The matrix block references identify the BWR operating state, the event number, and the system number. For example, F40-91 identifies BWR operating state F (Matrix 3), event (row) No. 40, and system (column) No. 91.

System Action

The SCS operates to limit the release of airborne radioactive materials to the environs.

Number Provided by Design

1. One Reactor Building

2. One main stack

3. One RBICS, with two dampers in series provided throughout the isolation system to provide redundancy. The control system is designed such that all dampers fail closed on loss of dc power to the solenoids or on loss of instrument air to the valves.

4. One SGTS consisting of two identical, parallel air filtration assemblies and two full capacity exhaust fans.

Minimum Required for Action

BWR Operating States A,B,C,D,E, and F:

The Reactor Building

(A40-90)	(B40-90)
(C39-90)	(D39-90)
(E39-90)	(F39-90)

The Main Stack

(A40-105)	(B40-105)
(C39-105)	(D39-105)
(E39-105)	(F39-105)

One damper at each isolation point (with associated controls)

(A40-102)	(B40-102)
(C40-102)	(40-102)
(E40-102)	(F40-102)

One filtration assembly train and one exhaust fan

(A40-91)	(B40-91)
(C39-91)	(D39-91)
(E39-91)	(F39-91)

5.3.7 Current Technical Specifications

The current limiting conditions for operation, surveillance requirements, and their bases are contained in the Technical Specifications referenced in Appendix B.

9.4 GASEOUS RADWASTE SYSTEM

9.4.1 Power Generation Objective

The Gaseous Radwaste System processes gaseous radioactive wastes from the main condenser air ejectors, the startup mechanical vacuum pump, the gland seal condensers, and other minor sources, and controls their release to the atmosphere through the main stack in such a way that the operation and availability of the station is not limited.

9.4.2 Power Generation Design Basis

1. The Gaseous Radwaste System is designed so that gaseous and particulate radwastes are processed and discharged such that operation and availability of the station is not limited.

2. The Gaseous Radwaste System is designed to minimize the possible explosion hazard of the hydrogen and oxygen present.

9.4.3 Safety Design Basis

1. The Gaseous Radwaste System is designed to include equipment, instrumentation, and operating procedures such that the gaseous radwastes can be discharged from the station at levels which are as low as reasonably achievable.

2. The Gaseous Radwaste System is designed to provide isolation on high offgas radioactivity level.

3. The Gaseous Radwaste System is designed to maintain its integrity for all expected operating conditions by conservative process design.

9.4.4 Description

9.4.4.1 Air Ejector Offgas and Augmented Offgas System

9.4.4.1.1 General

The Air Ejector and Augmented Offgas System shown on Figures 11.4-1 and 9.4-1 includes the subsystems that process and/or dispose of the gases from the main condenser air ejectors, the startup mechanical vacuum pump, and the gland seal condensers. All such gases from the unit are routed to the main stack for dilution and elevated release to the atmosphere. Discharges from the air ejector, the charcoal vault, and the stack are continuously monitored by radiation monitors.

Gases routed to the main stack include air ejector and gland seal offgases, and gases from the Standby Gas Treatment System (SGTS). Dilution air input to the stack is supplied by two full capacity fans located in the filter building at the base of the main stack. The stack is designed such that prompt mixing of all gas inlet streams occurs in the base to allow location of sample points as near the base as possible. The stack drainage is routed to the liquid radwaste collection system.

As a design basis for the system, a noble gas input equivalent to an annual average offgas release rate (based on 30 min decay, of 100,000 microcuries/sec was used. Table 9.4-1 indicates the design basis noble gas activity referenced to 30 min and the noble gas activity after processing through the Augmented Offgas System. Also shown on Table 9.4-1 are individual noble gas isotope activity reduction factors and the overall activity reduction factor provided by the system. The Augmented Offgas System receives the noncondensible gases discharged from the main condenser. These gases and their volumetric flow rates are given on Table 9.4-2. The air inleakage design basis for the Pilgrim Nuclear Power Station Augmented Offgas System has been established at 7 ft^3/min (at 130°F, 1 atm) per condenser shell. Leakage from two condenser shells (corrected to standard conditions) gives a total of 12.3 standard ft^3/min, the design basis air inleakage for the system.

Air inleakage at three operating boiling water reactors where condenser inleakage has a significant effect on offgas holdup time is given on Table 9.4-3.

9.4.4.1.2 System Function

The Augmented Offgas System shown on Figure 9.4-1 uses a high temperature catalytic recombiner to recombine radiolytically dissociated hydrogen and oxygen from the Air Ejector System. Noncondensible radioactive offgas is continuously removed from the main condenser by the air ejector during plant operation. The air ejector offgas normally contains activation gases, principally N-16, O-19, and N-13. The N-16 and O-19 have short half lives and quickly decay. The 10 min half-life N-13 is present in small amounts which is further reduced by decay. The air ejector offgas also contains the radioactive noble gas parents of biologically significant Sr-89, Sr-90, Ba-140, and Cs-137. The concentration of these noble gases depends upon the amount of tramp uranium in the coolant and on the reactor fuel cladding surfaces (usually extremely small) and the number and size of fuel cladding leaks. After hydrogen/oxygen recombination and chilling to strip the condensible to reduce the volume, the remaining noncondensibles, principally kryptons, xenons, and air, are delayed in a 30 min holdup system before reaching the adsorption bed. Radioactive particulate daughters of the noble gases are retained on the HEPA filters and on the charcoal. The charcoal adsorption bed, operating in a constant temperature vault, selectively adsorbs and delays the xenons and kryptons from the bulk carrier gas, principally air. This delay on the charcoal permits the Xe and Kr to decay in place. The offgas is discharged to the environs via the main stack. The activity of the gas leaving the Offgas Treatment System is continuously monitored. This system results in a reduction of the offgas activity (curies) released by a factor of approximately 185 relative to a 30 min holdup system as shown on Table 9.4-1.

The adsorption of noble gases on charcoal depends upon gas flow rate, holdup time, mass of charcoal and a gas unique coefficient known as the dynamic adsorption coefficient. The parametric inter-relationships and governing equations are well proven from three years of operation of a similar unit at KRB in Germany.

The design requirements for the equipment of the Offgas System are given on Table 9.4-4. The Augmented Offgas System is designated seismic Class II. This class includes those structures, equipment, and components which are important to reactor operation, but are not essential for preventing an accident which would endanger the public health and safety, and are not essential for the mitigation of the consequences of these accidents. A Class II designated item shall not degrade the integrity of any item designated Class I.

The front end components of the system are installed in the Turbine Building. The charcoal adsorbers and associated auxiliary equipment are installed in the Retention Building whose access doors are at elevation 23 ft. As described in Section 2.4.4.3, station structures at elevation 23 ft are not subjected to flooding.

The system is not designed to be functional during or after a tornado. The system is not essential for the prevention of accidents nor is it essential for the mitigation of the consequences of such accidents.

The Offgas System is provided with flow, temperature, and radiation instrumentation to ensure proper operation and control. Hydrogen analyzer instrumentation is also provided to ensure that hydrogen concentration is maintained below the flammable limit. Table 9.4-5 lists process system alarms and their location.

Offgas radiation monitoring is divided into two subsystems. One subsystem takes a continuous sample from the offgas line just after the air ejectors. The other takes a continuous sample from the Offgas System just before discharge to the main station stack. The former subsystem is described in Section 7.12.2.

The subsystem monitoring the Offgas System upstream of the main station stack has two instrumentation channels. Each channel consists of a gamma-sensitive detector, a pulse preamplifier, a logarithmic radiation monitor with a power supply and a meter, and a strip chart recorder point. The monitors and the two-pen recorder are located in the control room. Each logarithmic radiation monitor is powered from a different bus of the 24V dc system. The two gamma-sensitive scintillation detectors are mounted in two shielded sample chambers. The sample is drawn from the offgas line through the sample chamber by the sample pump. Each monitor has three upscale trips and a downscale trip. An upscale trip indicates high radiation. A downscale trip indicates instrument trouble. Any one trip will give an alarm in the control room. Any one upscale high radiation trip closes the charcoal bed filter bypass valve, if open, and opens the offgas line to the charcoal bed, if closed. Two upscale high-high radiation trips, one upscale high-high radiation trip and one downscale trip or two downscale trips isolate the Offgas System outlet and drain valves (See Figure 11.4-1). Subsequent operator action is required to reopen the valves. This will ensure positive control of releases to the environment.

The Offgas System radiation monitors have monitoring characteristics sufficient to provide accurate indication of radioactivity in the air ejector offgas, and provide the operator with sufficient information to monitor the performance of the Augmented Offgas System. Sufficient redundancy is provided to allow maintenance and testing of the Radiation Monitoring System. Each channel can be calibrated by analyzing a grab sample.

Figure 9.4-2 shows the location and arrangement of the front end components of the Augmented Offgas System in the Turbine Building. Figure 9.4-3 shows the arrangement of the charcoal adsorbers and | adsorber auxiliary equipment in a building located approximately 20 ft south of the Turbine Building between column lines 3 and 8. The building is approximately 68 ft by 72 ft in plan and extends 20 ft above and below grade.

9.4.4.1.3 System Operation

Noncondensible gas removed from the main condenser, including air inleakage, is diluted with steam to less than 4 percent by volume hydrogen concentration in the steam jet compressors. See Figure 9.4-1. The diluted offgas is superheated and then passed through a catalytic recombiner to convert the hydrogen and oxygen into water. The offgas effluent from the recombiner, containing only traces of hydrogen, is passed through a condenser, cooled by condensate, to remove the bulk moisture, and then to a 30 min holdup for the decay of the N-13, N-16, O-19, krypton and xenon isotopes. Decay daughters and iodine are removed by condensation on the walls of the holdup pipe and further removal of the decay daughters is effected by filtration. The offgas is processed by a cooler-condenser to remove additional moisture and iodine, a deentrainer and reheater to reduce the relative humidity, and a high efficiency filter prior to entering the charcoal adsorbers.

The charcoal adsorbers provide further delay of the xenon and krypton isotopes in the offgas. Two parallel trains of adsorbers are used to minimize back pressure. Heat is removed from the vault housing the adsorbers to maintain the charcoal beds at an approximate constant temperature of 77°F. The offgas effluent from the adsorbers is passed through another high efficiency filter prior to discharge to the offgas stack.

No dilution air is added to the offgas stream during steady state operation. The air present during operation is from air inleakage into the main condenser which operates at sub-atmospheric pressure. Oil free air is bled into the system during startup of the system. Its flow rate is 56.7 lb/hr, which is stopped after the recombiner comes up to temperature. Air is supplied during recombiner startup in order to prevent wetting of the recombiner catalyst and subsequent deterioration of the hydrogen recombiner performance.

In the event of failure of a nonredundant Augmented Offgas System component, provisions are made to bypass the Augmented Offgas System and operate the station using the installed 30 min Offgas Holdup System until maintenance of the Augmented Offgas System can be completed.

9.4.4.1.4 Safety Evaluation

The decay time provided by the Augmented Offgas (AOG) System permits significant radioactive decay of the activation gases and fission gases in the main condenser offgas prior to release. The design basis holdup is 22 days for xenon isotopes and 29 hr for krypton isotopes. The function of the AOG system is to limit offsite releases to provide assurance that releases of radioactive gaseous effluents will be kept As Low As Reasonably Achievable. Operation of the system with hold up times which are less than the design values will reduce the operational capability of the system but it does not result in a condition outside the analyzed bounds so long as the Technical Specification limits on release rate remain unchanged. Operation below the solid line in Figure 9.4-5 will provide assurance that release rates remain below Technical Specification limits and operational objectives. The ratio of Kr holdup time to Xe holdup time on Figure 9.4-5 is 1:17. The Steam Jet Air Ejector monitor is located prior to the AOG charcoal beds. The solid daughter products of the decay of the noble gases are removed by filtration and/or are retained on the charcoal. Final filtration of the charcoal adsorber effluent precludes escape of charcoal fines. Particulate activity release is expected to be negligible.

Iodine input into the offgas system is small because of its retention in reactor water and condensate. Additional iodine removal is provided by steam condensation which occurs in the offgas condenser located downstream of the hydrogen recombiner. Minute quantities of iodine entering the charcoal adsorbers are further adsorbed.

The Augmented Offgas System radiation monitors, which are normally monitoring the release rate from the adsorber beds, can be selectively valved to monitor the release rates from the recombiner outlet, the HEPA filters' outlet, or the outlets of the first charcoal bed in each train, all of which provide diagnostic information on the performance of the charcoal bed Holdup System. The charcoal bed adsorber radiation monitor also automatically isolates the Offgas System in the event of high radiation levels in order to prevent treated gas of unacceptably high activity from discharging to the atmosphere.

Shielding is provided for offgas system equipment to maintain safe radiation exposure levels for plant personnel. The equipment is principally operated from the control room.

The charcoal adsorbers operate in a massive temperature controlled vault of 77°F so that upon system shutdown, radioactive gases on the adsorbers will be subject to the same holdup time as during normal operation, even in the presence of continued air flow. The adsorbers are maintained at a constant temperature by an air conditioning system which removes the decay heat generated in the adsorbers. Failure of the Air Conditioning System will cause an alarm in the control room. In addition, a radiation monitor is provided to monitor the radiation level in the charcoal bed vault. High radiation will cause an alarm in the control room.

The hydrogen concentration of the gases from the air ejector is maintained below the flammable limit by maintaining adequate steam flow for dilution at all times. This steam flow rate is monitored and alarmed. The preheaters are steam heated rather than electrically heated in order to eliminate the presence of potential ignition sources and to limit the temperature of the gases in the event of cessation of gas flow. The recombiner temperatures are monitored and alarmed to indicate any deterioration of performance. A hydrogen analyzer downstream of the recombiners provides an additional check.

The Air Ejector Offgas System operates at a pressure of about 5 psig or less so the differential pressure which could cause leakage of radioactive gases is small. To minimize the possibility of leakage of radioactive gases, the system is welded wherever possible and bellows seal valve stems or equivalent are used.

Operational control is maintained by the use of radiation monitors to assure that the release rate is within the established limits. Environmental monitoring is used to determine resultant dose rates and to relate these to the release rates as a check on station performance. Provision is also made for sampling and periodic analysis of the influent and effluent gases for purposes of determining their composition. This information is used in calibration of the monitors and in relating the release to environs dose. The operator is thus in full control of the system at all times.

9.4.4.1.5 Malfunction and Failure Mode Analysis

Table 9.4-6 contains a detailed malfunction and failure mode analysis indicating the consequences of failure of various components of the system and design precautions taken to prevent such failures.

9.4.4.1.6 Inspection and Testing

In the Augmented Offgas System calibration and maintenance of monitoring equipment is performed on a routine basis. However, various temperatures, flow rates, and level signals are continuously monitored to detect for possible system malfunctions. Remote multiplexers transmit these signals, on demand, to a receiver in the main control room and on to the system computer. The computer calculates various parameters and sequences them for display on led readouts and on a multipoint trend recorder.

The particulate filters are tested after installation using a dioctylphthalate (DOP) smoke test equivalent. During operation, they will be periodically tested by laboratory analyses of inlet and outlet millipore filter samples.

Experience with boiling water reactors has shown that the calibration of the offgas and effluent monitors changes with isotopic content, isotopic leaks in the reactor, and the nature of the leaks. Because of this, the monitors are calibrated against grab samples periodically and at any time there appears to be a significant change in offgas release rate.

9.4.4.2 Turbine Sealing and Mechanical Vacuum Pump Systems

9.4.4.2.1 System Function

The Gland Seal Holdup System collects and processes by delay the noncondensible exhaust from the main turbine gland seal condenser. During startup operation the discharge of the condenser mechanical vacuum pump is routed through the Gland Seal Holdup System. The effluent of the Gland Seal Holdup System is routed to the main station stack where it is continuously monitored by the main stack Radiation Monitoring System before discharge to the environment. See Section 7.12.3.

During normal operation of the Gland Seal Holdup System, a 2,200 lb/hr saturated air-water vapor mixture containing trace amounts of hydrogen, oxygen, and radioactive gases is exhausted from the turbine generator gland seal condenser and enters the 16 in dia holdup line. After being delayed for a period of approximately 1.75 min, the effluent is routed to the main stack where it is mixed with the Augmented Offgas System effluent and the discharge of the stack dilution fans before release to the environment. See Figure 11.4-1.

9.4.4.2.2 System Operation

The Gland Seal Holdup System shares with the Augmented Offgas System the main stack, dilution fans and the stack Radiation Monitoring System.

During normal operation, the amount of radioactive activation and fission gases associated with the Gland Seal Holdup System is extremely small. The radioactivity that is collected and processed by the Gland Seal Holdup System is proportional to the amount of main steam utilized in the main Turbine Sealing System. This amount of steam is less than 0.1 percent of the full power rated steam flow. In addition to the small amount of radioactivity processed, there is a correspondingly small amount of radiolytic hydrogen and oxygen which are well below the explosive limits.

The Gland Seal System is designed to provide a 1.75 min holdup delay time for the radioactive gases before discharge to the main stack. This design is consistent with maintaining discharges within allowable limits due to the extremely small amount of radioactivity associated with this system.

During startup operations, the condenser mechanical vacuum pump is used to assist the steam jet air ejectors in achieving condenser vacuum. The discharge of the mechanical vacuum pump is routed through the Gland Seal Holdup System. The holdup normally provided by the Gland Seal Holdup System is reduced during startup due to higher air throughput when the mechanical vacuum pump is operating. Because the radioactive gases in the main condenser during startup are only a small fraction of the design evolution rate, the effect on radioactive effluents released to the environment is negligible.

The magnitude of the sources and resultant site boundary exposures resulting from station startups utilizing mechanical vacuum pump operation are difficult to quantify because the number, nature, and duration of the preceding shutdowns are difficult to estimate. An order of magnitude estimate of the annual average exposure from ten startups per yr was performed, assuming 4 hr of mechanical vacuum pump operation per startup, which indicated that maximum whole body exposures are less than approximately 0.05 mrem/yr. These estimated exposures can be reduced by minimizing the duration of mechanical vacuum pump operation.

9.4.4.2.3 Safety Evaluation

The amount of radioactivity associated with the Turbine Sealing System is negligible. The extremely low levels of radioactivity released from the Gland Seal Holdup System make direct radiation monitoring impractical; therefore, the total stack effluent is continuously monitored by the stack Radiation Monitoring System. Excessive release of radioactivity from this system is not considered credible due to its passive design and the small amount of main steam utilized in the sealing process.

The Gland Seal Holdup System is a passive system operating at atmospheric pressure requiring no particular control or instrumentation. Monitoring of the Gland Seal Holdup System effluent is provided by the stack Radiation Monitoring System which monitors the combined effluents of the Gland Seal Holdup System and Augmented Offgas System.

9.4.4.2.4 Inspection and Testing

The Turbine Sealing System is continously operated during station operation and does not require specific testing to ensure operability.

9.4.4.3 Miscellaneous Gaseous Effluents

9.4.4.3.1 Low Release Potential Effluents

Miscellaneous gaseous effluents are categorized into two classes, those from areas having a negligible or low potential for the release of airborne radioactivity, and those from areas likely to experience radioactive contamination. Following is a list of station areas which fall into these catagories. These areas are exhausted directly to the environment.

1. Diesel Generator Building

2. Administration Building

3. Machine Shop

4. Battery Room and Lube Oil Compartments

5. Recirculation Pump MG Set Area

6. Turbine Building Operating Floor and Switchgear Area

The ventilation air from the first five areas listed above has a negligible potential for the release of radioactive effluents. The Turbine Building operating floor including the reactor feedwater pump area are considered to have a low potential for release. Any release from the Turbine Building basement area or the Turbine Building ground floor to the Turbine Building operating floor or adjacent areas above elevation 51 ft is precluded since the Turbine Building basement and ground floor are maintained at a slight negative pressure relative to the Turbine Building operating floor.

The airborne radiation concentration levels at elevation 51 ft in the Turbine Building are routinely monitored by means of a constant air monitor with local alarm and recorder. Airborne activity levels in those areas of the station having a direct release path to the environs not monitored by a process radiation monitoring system will under normal operating conditions be within those levels allowed for in Appendix B, Table I, of 10CFR20, revised as of January 1, 1970. Assuming release from the Turbine Building operating floor at concentrations up to 3 X 10^{-9} microcuries/cm^3 the resulting concentrations at the site boundary would be less than 10^{-3} to 10^{-2} of MPC, on an unidentified basis.

The expected airborne activity on the Turbine Building operating floor will normally be below the values assumed above and the releases from the Turbine Building operating floor and the reactor feedwater pump area are expected to be insignificant relative to the releases from the main stack and the Reactor Building exhaust vent.

9.4.4.3.2 Potentially Contaminated Effluents

Gaseous effluents from areas of potential radioactive contamination are monitored and discharged to the environment through either the main stack or the Reactor Building exhaust vent. See Figures 10.9-4, 10.9-5, 11.4-1, and Section 10.9. The station ventilation systems are designed to combine the ventilation air flow from these areas and exhaust that air past process radiation monitoring equipment. The operation of the process radiation monitoring equipment is described in Sections 7.12.3 and 7.12.5.

Miscellaneous sources of potential low level radioactive airborne contaminants in the station which could be released to the environment are:

a. Primary Containment Venting

Primary system leakage inside the primary containment could occur as a result of recirculation pump seal leakage , valve flange leakage, and valve stem packing leakage. The latter two are also sources of potential leakage outside the primary containment. The magnitudes of these leaks are minimized to the extent possible by regular periodic inspections and station maintenance activities.

An analysis was performed to estimate the site boundary exposures resulting from primary containment purging assuming a 5-gal/min unidentified steam leak for a period of time sufficient to reach equilibrium concentrations in steam equivalent to a 25,000 microcurie/sec offgas rate after 30 min decay was used. The resultant whole body exposure per purge is estimated to be less than 0.001 mrem assuming that purging commences after the reactor has been brought to hot standby. During station operation, the drywell atmosphere is sampled for activity level to ensure that releases from this source will be minimal.

b. Steam Leakage Outside the Primary Containment

The site boundary exposures resulting from steam leakage outside the primary containment have been estimated based upon releases equivalent to a continuous steam leak of 7 gal/min of saturated liquid from the station ventilation exhaust. This has been selected based on experience at operating plants. The release to the environment may occur from the Turbine Building roof vent or the reactor building exhaust vent. Upper estimates of the magnitude of the whole body exposure resulting from noble gas releases from the steam range from 0.1 to 0.4 millirem/yr.

Assuming a leak rate of 7 gal/min and a coolant concentration concentration consistent with an offgas release rate of 25,000 microcuries/sec as measured at 30 min decay and a condensation plateout factor of 2 results in an environmental release rate of 0.04 microcuries/sec of I-131, with corresponding releases of I-132 to I-135. The value of 0.04 microcuries/sec release rate for I-131 can be compared to measurements which have been made on operating BWRs which have shown release rates from the building ventilation systems of 2×10^{-3} microcuries/sec to 4×10^{-2} microcuries/sec. The rate of release predicted results in a site boundary exposure rate of 0.6 millirem/yr.

c. Tank Vents and Sumps

Vents from liquid waste storage tanks, aerated resin regeneration tanks, open equipment, and floor drain sumps provide very little potential for contamination release, as the quantities of radioactive noble gases present in these liquids are generally negligible.

Particulate activity in the air space above stored liquid radwaste solutions is related to the gas liquid partition coefficient at the air water interface. The magnitude of this coefficient coupled with the filtration of a majority of the vents through high efficiency particulate (HEPA) filters minimizes the potential for particulate releases from liquid waste storage tanks and open sumps. The collection of ionic halogens on station demineralizer resins creates an additional potential source of noble gases through the decay of halogens to their noble gas daughters. However, these gases are not released promptly to the environment. The operation of the liquid radwaste system minimizes the release of gaseous daughter fission products by allowing the system components to act as gaseous delay tanks to effect the decay of the significant noble gas daughters. Only the occasional necessary air scrubbing and air sparging of certain radwaste tankage and normal tankage filling provides potential release mechanisms. The site boundary dose contribution from these sources is expected to be negligible.

d. Hood Vents

Radiochemical hood vents provide a potential miscellaneous source of release of airborne activity from the station. However, the sampling frequencies and volumes result in releases which are small fractions of the releases from other miscellaneous sources from the station. Further, the HEPA filters installed in the exhaust ducting from the radiochemical hoods act to ensure that no particulate radioactivity is released.

e. HPCIS Testing

The site boundary exposure due to testing of the High Pressure Coolant Injection System (HPCIS) for an assumed 30 hr/yr has been evaluated. The HPCIS turbine uses primary system steam for motive force of which 500 lb/yr is used as HPCIS turbine gland sealing steam and is condensed in the HPCIS gland seal condenser. The associated noncondensibles including trace amounts of noble gases are released during test operation to the environment through the reactor building exhaust vent or the Standby Gas Treatment System (SGTS) which discharges to the main stack. The resultant site boundary whole body exposure is negligible less than approximately 1 percent of that expected due to primary system leaks outside the primary containment.

9.4.5 Estimates of Radioactive Gaseous Releases During Normal Operation

Estimates of radioactive gaseous releases and resultant doses during normal operation are given in the Pilgrim Station Unit 1 Appendix I Evaluation, dated April 1977.

10.9 HEATING, VENTILATION, AND AIR CONDITIONING SYSTEMS

10.9.1 Power Generation Objective

The power generation objective of the Heating, Ventilation, and Air Conditioning (HVAC) systems is to control the station air temperatures and the flow of airborne radioactive contaminants to ensure the operability of station equipment, and the accessibility and habitability of station buildings and compartments.

10.9.2 Power Generation Design Basis

The HVAC systems:

1. Provide temperature and humidity control and air movement for personnel comfort and optimum equipment performance
2. Provide a sufficient filtered fresh air supply for personnel
3. Provide for air movement from lesser to progressively greater areas of radioactive contamination potential prior to final exhaust
4. Minimize the possibility of exhaust air recirculation into the air intake

10.9.3 Description

10.9.3.1 General

The HVAC systems provide individual air supplies to main areas of the station. Normal airflow is routed from lesser to progressively greater areas of radioactive contamination potential prior to final exhaust.

Most of the heating and ventilating systems utilize 100 percent outside air with no recirculation. All inlet air can be temperature controlled with heating coils.

Air from areas containing potential sources of radioactive contamination such as the Reactor Building, Radwaste Building basement, Turbine Building basement, and Radwaste Heating and Ventilation Building are discharged through the building exhaust vent. Air from other areas is discharged at building roof levels.

The Equipment Area Cooling System (EACS) is described in Section 10.18. Main control room heating, ventilation, and environment control are described in Section 10.17.

10.9.3.2 Station Heating System

The station is heated during plant operation by a forced circulation, hot water system. See Figures 10.9-1 and 10.9-2. The system consists of two package boilers, five hot water circulating pumps, two fuel oil transfer pumps, two 25,000 gal fuel storage tanks, one compression tank and associated piping, valves, combustion controls, and instrumentation.

Heating system water is heated by two oil-fired package boilers. The boilers are located in the Reactor Building auxiliary bay. The hot water boilers are each rated at 16.7 X 10^6 Btu/hr at 125 psig, utilizing No. 2 diesel fuel oil.

The station heating system utilizes a low temperature (210°F max.), two pipe, forced circulation hot water system. System water temperature is automatically controlled based on outside air temperature. Water at temperatures adequate to maintain space temperatures is provided to unit heaters located throughout the station.

Three main system circulating pumps, each sized for one-half of the total design flow rate, provide continuous flow throughout the system. Normally, two run with one standby. Two booster pumps sized for full Turbine Building hot water coil capacity will provide additional capacity during low temperature periods. One pump runs with one pump a standby. The main heating piping headers are cross connected to allow isolation of either boiler when their use is not required. The auxiliary boilers and circulating pumps will be accessible for maintenance.

Heating system water temperature control is provided by a three-way mixing valve located in the hot water supply line on the suction side of the heating system circulating pumps. The actuating device for the three-way valve is an outdoor air sensing element controlling the position of the valve mixing unit to blend boiler supply and return water. This varies system water temperature inversely with outdoor temperature to reduce fluctuations in space temperatures as demand varies. During mild weather periods the boilers have a reduced flow; however, the heating input is reduced proportionately. The auxiliary boiler water temperature is controlled by a temperature insertion element controlling a modulating oil burner metering assembly.

Heating and ventilating equipment hot water coils have constant flow during pump operation. Leaving air temperature is controlled by an air temperature sensor modulating face and bypass air dampers.

Air conditioning fan unit hot water coils leaving air temperatures is controlled by a three-way valve located in the return line. Unit heaters have constant flow through their coils at all times during pump operation. Space temperatures are controlled by a space thermostat controlling the unit heater fan motor.

Winter design temperatures for the system are given on Table 10.9-1 and summer design temperatures are given on Table 10.9-2.

10.9.3.3 Reactor Building Heating and Ventilation System

The Reactor Building is divided into three major ventilation zones. One zone encloses the spaces above the operating (refueling) floor. The second zone encloses the recirculation pump motor generator sets and the third zone encloses the remainder of the Reactor Building. Each zone is served by its own air supply and exhaust system in order to maintain the independence of the zones. See Figure 10.9-3. The systems for the first and third zones employ once-through ventilation without recirculation. The design basis summer maximum temperatures are area specific (See Table 10.9-2).

The operating floor ventilation zone is normally supplied with 30,000 ft^3/min of filtered and tempered outside air which enters the Reactor Building through louvers in the east wall. Two supply fans, each rated at design capacity, are located in the Reactor Building in a fan room. The fans are manually started. For normal operation one fan runs while the other fan is on standby. However, both fans can be operated in parallel to provide additional ventilation during refueling operations. If the operating fan fails during normal operations, the standby fan starts automatically and an alarm is received in the main control room. Air is exhausted from the operating floor through ducts located in the roof truss area and the south wall area adjacent to the floor. Additional exhaust ducts are located above the water level in the fuel pool, steam dryer, and separator pool, and in the reactor cavity. Two operating floor exhaust fans, each rated at design capacity, are located in the main fan room outside the Reactor Building. These fans discharge into the main exhaust plenum at the base of the building vent. The building vent is square in cross section extending from the top of the main exhaust plenum to the discharge point at elevation 182 ft msl.

The zone enclosing the recirculation pump motor generator sets is normally supplied with 50,000 ft^3/min of filtered outside air or recirculated air. Outside air enters through louvers in the west wall. Two supply fans, each rated at design capacity, are manually started. For normal operation one fan runs with the other as standby. Both fans can be operated in parallel to provide additional ventilation if required. If the operating fan fails, the standby fan starts automatically, and an alarm is received in the main control room. Two exhaust fans, each rated at design capacity, recirculate or exhaust zone air. Exhaust air is discharged through louvers in the north wall. Temperature control is provided by an air temperature sensor modulating the supply, exhaust, and recirculation dampers. Supplementary cooling for summer conditions is provided by unit coolers supplied by the Reactor Building Closed Cooling Water (RBCCW) System. Four unit coolers, each rated at half design capacity, are installed. Unit heaters are installed to provide zone heating during winter shutdown conditions. The heaters are supplied by the station heating system.

The remainder of the Reactor Building is supplied with a total of 60,000 ft^3/min of filtered and tempered outside air which enters the Reactor Building through louvers in the east wall into two fan rooms. Each fan room contains two supply fans rated at design capacity. Each fan is capable of supplying 30,000 ft^3/min of air to various reactor building areas. The fans are manually started, and for normal operation, one fan in each fan room runs while the other fan is at standby. If the operating fan fails during normal operations, the standby fan starts automatically, and an alarm is received in the main control room.

The air supplied to the Reactor Building from the two operating supply fans is routed through the building to areas of progressively higher contamination potential. Air exhausted from areas of higher contamination potential (contaminated area exhaust) is routed independently of the exhaust from areas of expected lesser contamination (clean area exhaust).

Two contaminated area exhaust fans, each rated at design capacity, are located in the Reactor Building. The fans discharge to the main exhaust plenum at the base of the building vent. An additional smaller exhaust fan located in the Reactor Building, exhausts only from the control rod drive maintenance shop, and discharges to the main exhaust plenum. Constant volume control is maintained by inlet vanes which are automatically positioned.

All the Reactor Building supply fans are electrically interlocked with their corresponding exhaust fans and run only when their associated exhaust fans are operating.

10.9.3.4 Turbine Building Heating and Ventilation System

The Turbine Building air flow diagram is shown on Figure 10.9-4.

10.9.3.4.1 General

The Turbine Building Ventilating System supplies filtered air to all areas of the Turbine Building and is routed to areas of progressively greater radioactive contamination potential prior to final exhaust.

The ventilation system supplies filtered and tempered outdoor air to the operating floor and all other areas below the operating floor. The main condenser area is maintained at a slightly negative pressure to prevent the spread of radioactive contaminants to the adjacent operating areas.

The exhaust air from the Turbine Building operating floor and switchgear areas is discharged to the atmosphere through exhaust fans located on the Turbine Building roof. The exhaust air from the reactor feedwater pump area will be discharged to atmosphere through exhaust fans located on roof above. The exhaust air from the condenser compartment and other adjacent potentially contaminated areas will be discharged through the building vent. The exhaust air from the battery rooms and lube oil compartments will be discharged by independent exhaust fans located on the Turbine Building operating floor through ductwork to the Turbine Building roof.

The condenser and condensate pump compartments depend on fan coil cooling units to supplement the main ventilation system when outside air temperatures are above 60F.

10.9.3.4.2 Main Turbine Building Ventilation Supply Fans

There are three Main Turbine Building ventilation supply fans, each of half capacity. Under normal operating conditions two fans are running, with one at standby. The fans are shut down in the event of loss of offsite power. During normal operations, any two fans are started. If an operating fan fails, a flow switch will sense a reduction of pressure and annunciate in the main control room. The standby fan is then started manually. If the fan discharge air temperature drops below 40F, a temperature switch will stop the fans and annunciate in the main control room.

Heating coils are provided to heat the outside supply air when necessary. The supply air temperature is controlled by modulating the amount of air flow over the heating coils with face and bypass dampers. Hot water flow to the heating coils is constant. The supply air volume is constant, with no recirculation. Outside air inlet dampers are interlocked with each supply fan and close if all fans are off. Outside air dampers are heated with resistance cable to prevent freezing. Constant volume is maintained by inlet vanes which are positioned automatically.

10.9.3.4.3 Turbine Building South Wall Ventilation Louvers (Operating Floor Level)

There are 3 louvered openings in the south wall of the Turbine Operating Floor. Each is 5' x 6', and they were originally designed to provide additional cooling for the operating floor during unusually warm weather. The unit on the west side has been permanently blocked off and the other two dampers are normally closed but can be manually opened if needed.

10.9.3.4.4 Turbine Building Basement Exhaust Fans (Condenser Compartment)

There are three half capacity Turbine Building basement exhaust fans which discharge to the building vent. The fans are shut down in the event of loss of offsite power. The fans are started manually. If an operating fan fails, a flow switch will sense a reduction of pressure and annunciate in the main control room. The operator will then start the standby fan manually. Constant volume control is maintained by inlet vanes which are automatically positioned. The air volume exhausted by the fans is sufficient to maintain a negative pressure relative to all adjacent areas.

10.9.3.4.5 Turbine Building Operating Floor Exhaust Fans

Nine direct drive fans (two are spares) are located on the roof, and seven run during normal operation. The fans are shut down in the event of loss of offsite power. The fans are manually started. Fan motor malfunction is annunciated in the main control room.

10.9.3.4.6 Lube Oil and Battery Room Exhaust Fans

Two full capacity fans are installed which exhaust to the Turbine Building roof. The fans are shut down in the event of loss of offsite power. The fans are manually started. If the operating fan fails, a flow switch will sense a loss of pressure and automatically start the standby fan and annunciate in the main control room. In order to detect fire damper failures, additional flow switches are provided to sound an alarm in the control room upon loss of flow in each Battery Room Exhaust System. Constant volume control is automatically maintained by the inlet vanes.

10.9.3.4.7 Condensate Pump Room Unit Coolers

Two full capacity fan unit coolers are installed. The units are shut down in the event of loss of offsite power. The fans are manually started. If the operating fan fails, a flow switch senses reduction of pressure, automatically starts the standby unit, and annunciates in the main control room. A temperature element (with local transmitter) is located in the discharge of each unit to give temperature indication in the main control room. Cooling water supply is from the Turbine Building Closed Cooling Water (TBCCW) System.

10.9.3.4.8 Condenser Compartment Unit Coolers

Six unit coolers (two are standby) are installed in the condenser compartment and are supplied with cooling water from the TBCCW System. The fans are started manually. High compartment temperature will alarm in the main control room and a standby unit is manually started. The leaving air temperature is adjusted by regulating the flow of cooling water to each unit cooler.

10.9.3.4.9 Off-gas Retention Building

The off-gas Retention Building is supplied 5400 CFM of air from the Turbine Building HVAC System and 100 CFM of air from process air systems. This is all exhausted back to the Turbine Building HVAC System (refer to Figure 10.9-5).

<u>Off-gas Retention Building Unit Coolers</u>

Three 2 horsepower unit coolers handle air received by the Retention Building. One cooler provides 3060 CFM to the operating floor and control room and utilize the TBCCW System as a heat sink. The other two (one is standby) provide 2340 CFM to the charcoal vault (refer to Figure 10.9-5).

Off-gas Retention Building Exhaust Fans

Two 5500 CFM exhaust fans (one is standby) draw air from the charcoal vault filters, equipment room, operating floor, process vents and drains, and other building sources to the Turbine Building HVAC System (refer to Figure 10.9-5).

10.9.3.4.10 Off-gas Retention Room Unit Coolers

Each Off-gas Recombiner Room contains a 5000 CFM unit cooler which circulates and cools room air. These units utilize the TBCCW System as their heat sink.

10.9.3.5 Radwaste Building Heating and Ventilating System

Radwaste area air flow diagram is shown on Figure 10.9-5.

10.9.3.5.1 General

The Radwaste Building Heating and Ventilating System maintains required space temperatures, provides adequate ventilation to remove heat rejected from operating equipment compartments, and provides adequate supply and exhaust to maintain the direction of air flow from lesser to increasingly greater areas of potential radioactive contamination. Exhaust hoods with negative pressure are provided at locations where, under normal operation, contaminants could escape to the surrounding areas. A pipe manifold vent system is provided for expected contaminated tanks and equipment. Filtered and tempered air is supplied to all personnel occupancy areas, the monitor tank compartment, and treated waste holdup tank areas in sufficient quantity to maintain space temperatures and ventilation. The air is supplied through ductwork by two full capacity air handling units, (one is normally running and one is a standby).

Exhaust air from the contaminated equipment tank cells, the vent pipe manifold, and the baling machine area, and other ventilation system exhausts are routed through the two banks of exhaust air filter assemblies and discharged to the building vent.

10.9.3.5.2 Radwaste Heating and Ventilating Units

Each of the two heating and ventilating units is full capacity. The fans are shut down in the event of loss of offsite power. The fans are manually started. If the normal operating fan fails, a flow switch senses a reduction of pressure and automatically starts the standby fan and annunciates in the main control room. The fan discharge air temperature is controlled by a thermostat through a limit sensor and controller which modulates the heating coil face and bypass air damper. The heating coil is supplied by the station hot water heating system. The outside air damper is heated with a resistance cable to prevent freezing.

10.9.3.5.3 Radwas : Exhaust Fans and Filters

The two exhaust fans are each full capacity. The fans are shut down in the event of loss of offsite power. The fans are manually started. If the normal operating fan fails, a flow switch senses loss of pressure, automatically starts the standby fan, and annunciates in the main control room. Constant volume is maintained by inlet vanes automatically positioned. Two half capacity filter assemblies are installed. Each filter assembly can be isolated manually. Prefilter and HEPA filter differential pressures are indicated outside the filter assembly compartment.

10.9.3.6 Access Control Area Air Conditioning

10.9.3.6.1 General

The access control area air conditioning system maintains ventilation and constant temperature and humidity in the access control area. See Figure 10.9-6. The equipment, ductwork, and controls are completely independent from other station HVAC systems. The system independence will insure uninterrupted operation during normal and shutdown modes.

The access control area is served by full capacity redundant units including dual duct (hot-cold) air conditioning units, reciprocating condensing units, and recirculation and exhaust fans.

Conditioned air is distributed through ductwork to the mixing boxes and diffusers located in the various zones. The zones independent from each other are:

Corridor

Instrument Repair

Chemical Laboratory and Counting Rooms

Frisking Area

Decon Shower

Dressing Area

H.P. Office

Chemical Lab Expansion

Undressing Area

Air is recirculated from the instrument repair room only. All the other rooms will be kept at a positive pressure with respect to the chemistry laboratory, thus allowing air to infiltrate from the lobby in the upper level into the chemistry laboratory. The exhaust from the laboratory fume hoods is exhausted by a booster fan through the radwaste filtering system prior to release from the building vent.

10.9.3.6.2 Access Control Area Air Conditioning Units

Two full capacity units are installed. The units are shut down in the event of loss of offsite power. The fans are manually started. If the normal operating fan fails, a flow switch will sense a loss of pressure, automatically start the standby fan, and alarm in the main control room. One full capacity air conditioning unit is provided for the H.P. Office area on El. 37'-0". The unit is shut down in the event of loss of offsite power. The unit is manually started and stopped. Space temperatures will be controlled by electric duct heaters. Each room is provided with its own thermostat for individual control.

10.9.3.6.3 Access Control Area Recirculation and Exhaust Fans

Two full capacity recirculation and two full capacity exhaust fans are installed The fans are manually started and are shut down in the event of loss of offsite power. The exhaust fans discharge to the building vent. The recirculation fans either recirculate the air or discharge to the environment at roof level. If the normal operating fan fails, a flow switch senses loss of pressure, automatically starts the standby fan, and alarms in the main control room. Portions of the air exhausted from the counting room and H.P.

counting work space is discharged into hot machine shop on El. 23'-0". The remainder of the air is returned to the air conditioning unit, mixed with outside air and redistributed to the space. The exhaust fan will shut down on loss of offsite power.

10.9.3.7 Intake Structure

10.9.3.7.1 General

The intake structure is comprised of five heating and ventilating areas: one area containing the salt service water pumps; two areas containing the condenser circulating water pumps; an area containing the fire pumps; and an area containing the hypochlorite storage tank. See Figure 10.9-6.

Exhaust fans are provided for the ventilation of each area. The salt water service pump area has redundant fan units while the other areas are served by single fans. Unit heaters are provided for heating. Air is supplied to the various areas from outside louvers. Outside air dampers are heated with resistance cable to prevent freezing of controls.

10.9.3.7.2 Service Water Pump Exhaust Fans

Two full capacity fans are installed and either fan will provide the required air flow for the pump rooms. The fans may be powered by the diesel generators. One fan is manually started, and should the operating fan fail, a flow switch senses loss of pressure, automatically starts the standby far and annunciates in the main control room. Constant volume control is maintained by automatically positioned inlet vanes.

10.9.3.7.3 Other Area Exhaust Fans

One full capacity fan is installed for each of the remaining intake structure areas. The fans are manually started and are shut down in the event of loss of offsite power.

10.9.3.8 Warehouse and Machine Shop

10.9.3.8.1 General

Filtered and temperature controlled air is supplied to the tool room, machine shop, and warehouse. See Figure 10.9-6. The air is supplied through ductwork by one air handling unit. Approximately 20 percent of the total supply air is exhausted through the exhaust hood over the decontamination trough, then into the radwaste air filtering unit. The remainder is exhausted to atmosphere through louvers at roof level.

The equipment, ductwork, and controls are completely independent from other HVAC Systems except for the decontamination exhaust to the radwaste area.

10.9.3.8.2 Heating and Ventilating Unit

One full capacity unit is installed. The fan unit is manually started and is shut down in the event of loss of offsite power. Fan failure is annunciated in the main control room. The supply air temperature is controlled by modulating face and bypass air dampers. Hot water flow to heating coils is supplied by the station heating system. Supply air flow is straight through with no recirculation. The outside air inlet dampers are interlocked with the supply fan and closed when the fan stops. Outside air dampers are heated with resistance cable to prevent freezing of controls.

10.9.3.8.3 Exhaust Fans

Full capacity fans exhaust the warehouse and machine shop areas and the machine shop decontamination trough. The fans are started manually and shut down in the event of loss of offsite power. If the area fan fails, a flow switch will sense a reduction of pressure and annunciate in the main control room.

10.9.3.9 Diesel Generator Building Heating and Ventilation

Each standby diesel generator room is heated by hot water unit heaters supplied by the Station Heating System. See Figure 10.9-3. Heating units are automatically shut down after the diesel generators start.

Outdoor air is supplied to each diesel generator room through separate air intake plenums. Dampers control the air flow from the plenums when a diesel generator is operating as follows:

1. Outside Air Damper - This damper is associated with the air flow path which removes heat from the Diesel Jacket Water Cooling System during operation. The Diesel Generator's radiator fan draws air from a common intake plenum (which also serves the Ventilation Damper described below) through the Outside Air Damper which is fully opened during Diesel Generator operation. The spent cooling air exhausts from the building through the Exhaust Damper (also fully open) & exhaust louvers located near the roof. Both the Outside Air and Exhaust Dampers are fully closed during shutdown. Neither damper has a manual override and both fail open upon loss of motive power or control signal to assure that cooling air is available.

2. Ventilation Air Damper - This damper is associated with the air flow path that provides general building ventilation as well as supplying the Diesel Generator engine with combustion air. It is also sometimes referred to as the Combustion Air Damper. In shutdown mode, the Ventilation Air Damper is shut. Upon initiation of the engine start signal, this damper fully opens to provide greater air throughout to support the engine vacuum. The Ventilation Damper has no manual override and fails open upon loss of motive power or control signal in order to assure there is always adequate combustion air available.

Engine freeze protection is provided by the Jacket Water Cooling System which maintains the engine coolant temperature at a constant value during shutdown mode. Moreover, the engine coolant contains antifreeze in the event that coolant cannot be heated during shutdown mode.

Ventilation system ducts, dampers, fan, and controls are Class I design.

10.9.3.10 Administration Building Air Conditioning

The Administration Building is supplied by a hot and cold duct air-conditioning unit with air returned or exhausted by a recirculation fan. Conditioned air is distributed through ductwork to zone mixing boxes and ceiling diffusers.

During the winter, hot water baseboard convectors located along the outside walls help maintain the required temperature inside the building. Zone temperatures are controlled by a thermostat in each zone. The leaving air temperature of the heating and cooling coils is preset with conditioned air flowing to zone mixing boxes. Zone thermostats control metering valves in each mixing box to control air temperature. Base board convectors are controlled independently by a thermostatic valve in the return line to each zone.

The Administration Building Air Conditioning uses the TBCCW System as a heat sink.

10.9.3.11 New Administration and Service Building HVAC

The New Administration and Service Building HVAC System is independent to those of the process buildings.

TABLE 10.9-1

DESIGN TEMPERATURES (WINTER)

Outdoor: 10°F Dry Bulb

Indoor:	Minimum
Turbine Building	60°F
Reactor Building	60°F
Access Control Area	75° FBD 50% relative humidity
Administration Building	75° FBD 50% relative humidity
Radwaste Area	65°F
Diesel Generator Building	60°F
Intake Structure	60°F
Machine Shop & Warehouse Area	65°F
Fire Water Storage Tanks	45°F
Demin. Water & Condensate Storage Tanks	45°F

TABLE 10.9-2

DESIGN TEMPERATURES (SUMMER)

Outdoor: 88°F Dry Bulb
74°F Wet Bulb

Indoor:	Maximum
Turbine Building	105°F - 120°F
Reactor Building	See Note (1 Below)
Control Room Area	78°FDB 50% relative humidity
Access Control Area	78°FDB 50% relative humidity
Administration Building	78°FDB 50% relative humidity
Radwaste Area	100°F
Diesel Generator Building	105°F (Max)* - 95°F (5ft above Floor)*
Intake Structure	105°F
Machine Shop and Warehouse Area	105°F
Cable Spreading Room	-102°F to +76°F

NOTE 1: The following data represent the Reactor Building maximum summer design temperatures by specific location:

Location	Max. Room Temp. (°F)	Av. Room Temp (°F) 5 Feet Above Floor Level
Refueling Floor	105	95
General Floor Area	105	95
Main Steam Pipe Tunnel	120	110
RHR/Core Spray Pump Area	115*	100*
CRD Pump Area	115	100
RCIC Pump Area	115*	100*
HPCI Pump Area	115*	100*
Cleanup Regn. & Nonregn. Heat Exchanger	115	100

* These temperatures apply only to conditions when the equipment in the area is operating. Under normal plant operation, temperature in these areas will be lower than indicated above.

PILGRIM NUCLEAR POWER STATION

SEMIANNUAL REPORT #1

ENVIRONMENTAL RADIATION MONITORING PROGRAM

July 1-December 8, 1972

Prepared for

The Boston Edison Company
800 Boylston Street
Boston, Massachusetts 02199

Prepared by

Environmental Analysts Incorporated
224 Seventh Street
Garden City, New York 11530

January 1973

ABSTRACT

Pilgrim Unit 1, a 1,998-MWt boiling water reactor station, began initial operation in June 1972. During this six-month start-up and testing period, reactor operation was intermittent and power levels were quite variable. Radionuclide discharges in the liquid and gaseous effluents were very low during this period.

The operational environmental radiation surveillance program is being conducted in accordance with the specifications in the Pilgrim Station operating license. In addition to the environmental monitoring program outlined in the license, Boston Edison maintained surveillance at other locations during this initial reporting period. No increases in environmental radioactivity levels outside the station property were evident in the monitoring data. The only statistically significant increase in radiation levels during this period was the measured gamma exposure rate at the overlook area TLD monitoring station, which is located approximately 400 feet from the turbine building. The measured exposure rate increased approximately 10 microroentgens per hour above background at this location in November. Access to this location is controlled by Boston Edison Company; however, a small visitors' center is open to the public from April through November. The average increased dose rate to the attendant was estimated to be from 0.6 to 0.8 millirem during this reporting period, and the average dose to individual visitors was estimated to be from 0.0004 to 0.003 millirem during October and November. The population dose to visitors at the overlook area during

October and November was estimated to be from 0.004 to 0.007 man-rem. Based on these limited measurements the annual population dose to visitors at the overlook area was estimated to be approximately 0.2 man-rem. More reliable dose estimates will be possible in the next semiannual report after additional measurements have been made.

I. INTRODUCTION

This is the first semiannual report of the Pilgrim Nuclear Power Station operational environmental radioactivity monitoring program, covering the period from July 1, 1972, to December 8, 1972. The data are reported in accordance with the operational environmental monitoring and reporting requirements of Appendix A to the Pilgrim Facility Operating License DPR-35, Technical Specifications and Bases.

Pilgrim Unit 1, a 1,998-MWt boiling water reactor, began initial start-up operations in June 1972. Over this initial six-month period the reactor power level was increased to approximately 98 percent of full power in November. The gradual power ascension and frequent outages occurred as a result of performing various required tests of component systems. Figure 1 shows the daily reactor thermal power levels from July through December. Radionuclide emissions in the liquid and gaseous effluents were minimal during this initial period of testing.

The monitoring program has been executed by Boston Edison Company personnel with analytical services provided by Interex Corporation (formerly ICN/Tracerlab).* Extensive measurements of background radiation were made during the preoperational monitoring program and serve as a baseline for evaluating the measured radioactivity levels during the operational program. The preoperational program began in 1969 and was also performed by Boston Edison personnel and Interex Corporation. Boston Edison personnel collected the environmental samples, except for marine life, and delivered them to Interex. Interex has performed the radioactivity

*1601 Trapelo Road, Waltham, Massachusetts.

measurements and has provided the results of analyses quarterly to Boston Edison. Marine life samples have been collected by the Massachusetts Division of Marine Fisheries and provided to Boston Edison Company for analysis. The thermoluminescent dosimeters (TLD) are changed monthly and read by Boston Edison personnel using a TLD reader at Interex Corporation.

In addition to the monitoring program defined in the operating license technical specifications, Boston Edison operated four other air sampling stations and seventeen additional TLD monitoring locations during this report period. This report includes the results obtained in the specified monitoring program and at the supplementary locations.

Interex Corporation had not completed all of the required analyses of environmental samples collected through December 8, 1972. The available results are reported in Section III of this report, and the results for incomplete analyses will be provided in the next semiannual report.

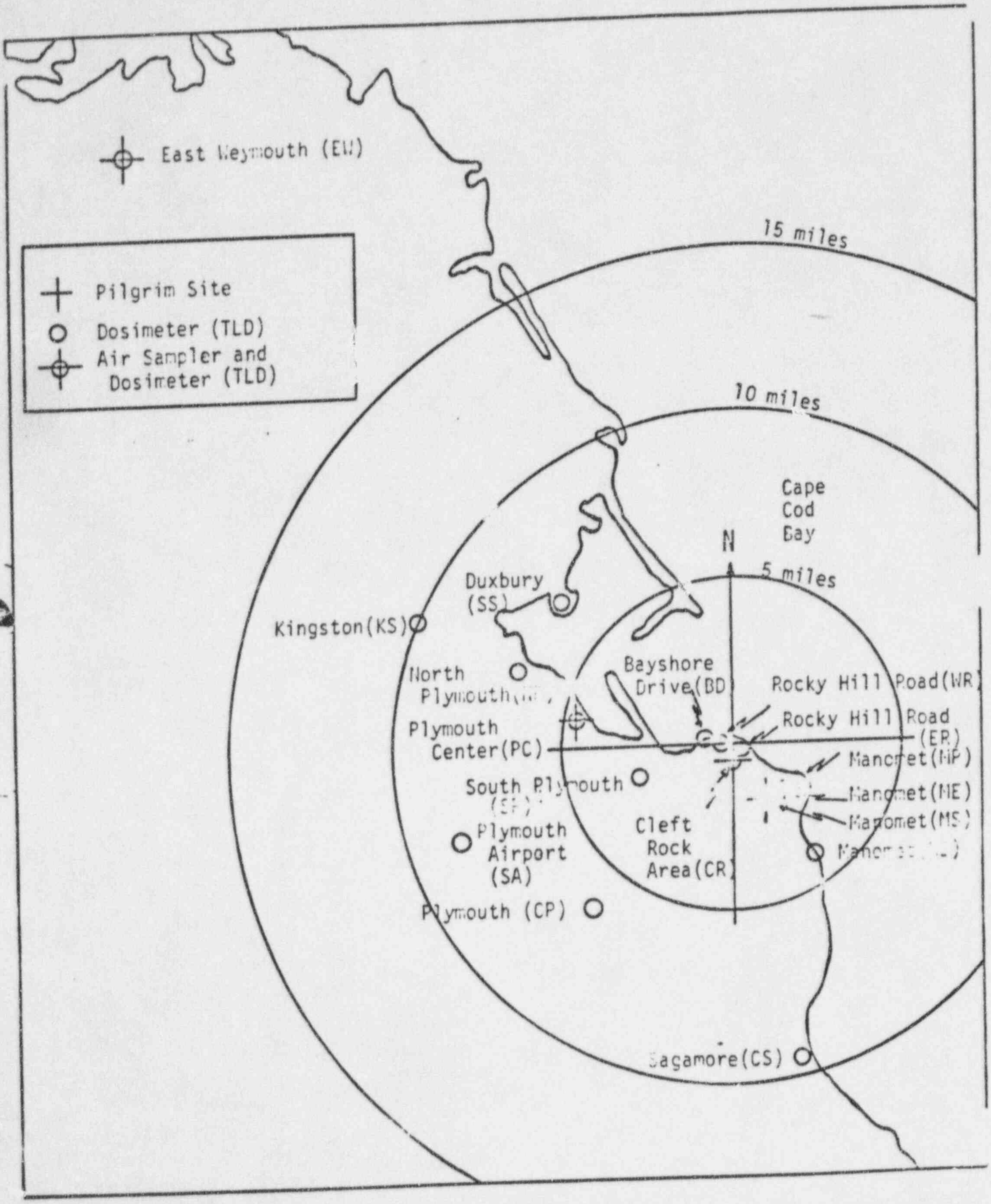

Figure 2. [illegible] Exposure and Air Particulate Monitoring Stations.

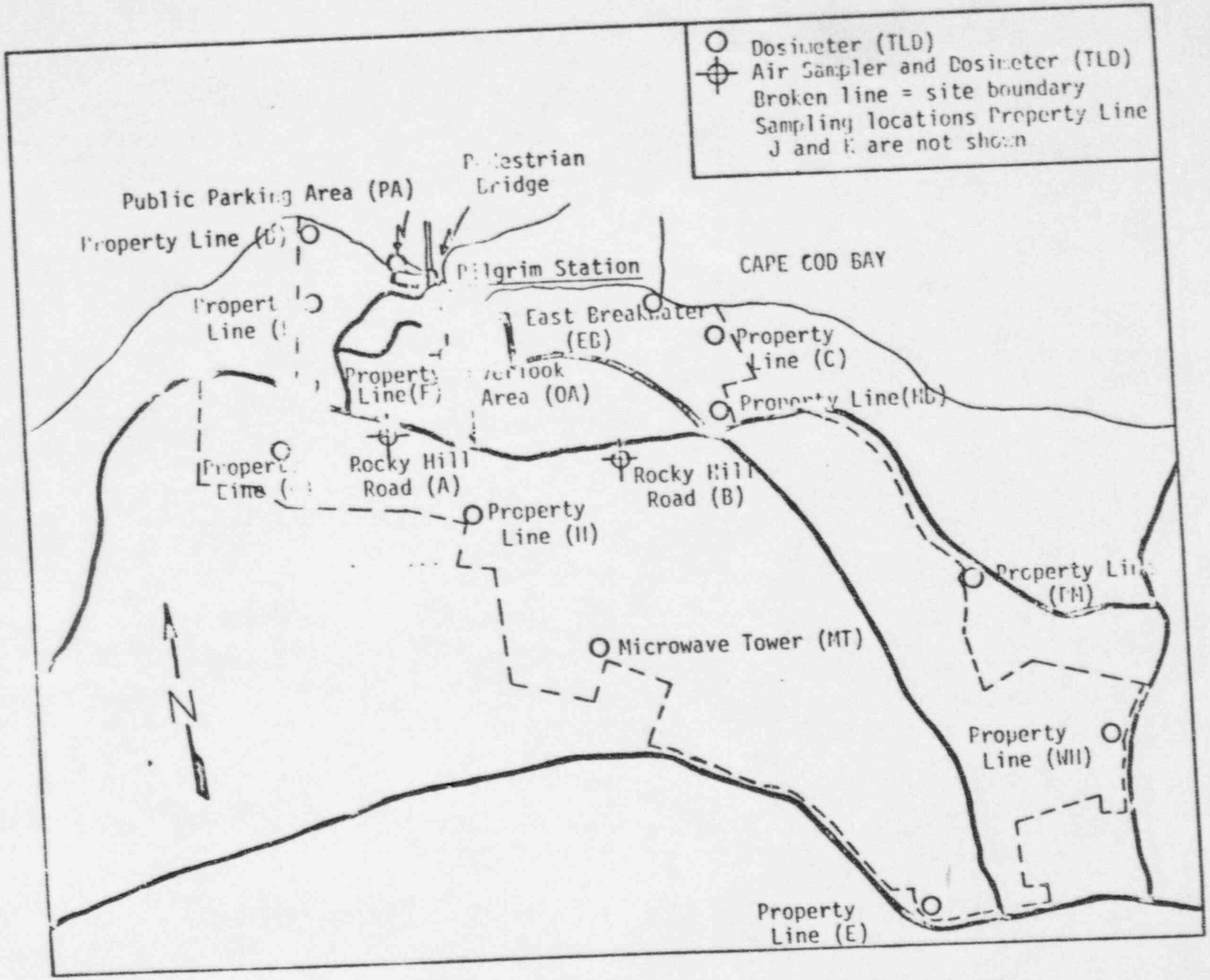

Figure 3. Locations of Onsite Exposure and Air Particulate Monitoring Stations.

Table 6

Air Particulates--Gross Beta Concentrations in Weekly Samples (pCi/m^3)[a]

Collection Period (197.)	Control	Offsite			Onsite						
	East Weymouth	Plymouth Center	Cleft Rock	Manomet Substation	West Rocky Hill Road	Overlook Area	Pedestrian Bridge	East Rocky Hill Road	Property Line	East Breakwater	Warehouse Sampler
July 6-July 13	0.23	0.26	0.18	0.18	0.22	0.16	0.22	0.18	0.19	0.13	0.20
July 13-July 21	0.06	0.04	0.01	0.04	0.04	0.04	0.05	0.04	0.04	0.04	0.04
July 21-July 27	0.11	0.14	0.13	0.12	0.13	0.10	0.13	0.11	0.14	0.11	0.12
July 27-Aug. 3	0.11	0.11	0.10	0.11	0.11	0.13	0.13	0.11	0.10	0.12	0.12
Aug. 3-Aug. 10	0.11	0.09	0.09	0.09	0.09	0.12	0.09	0.10	0.11	0.08	0.08
Aug. 10-Aug. 17	0.10	0.09	0.09	0.08	0.07	0.08	0.07	0.10	0.08	0.07	0.08
Aug. 17-Aug. 24	0.10	0.09	0.08	0.14	0.08	0.07	0.08	0.11	0.11	0.09	0.08
Aug. 24-Aug. 30	0.07	0.06	0.13	0.03	0.04	0.05	0.07	0.02	0.06	0.08	0.05
Aug. 30-Sept. 7	0.07	0.07	0.06	(b)	0.07	0.05	0.07	0.08	0.06	0.12	(b)
Sept. 7-Sept. 14	0.08	0.07	0.08	0.08	0.07	0.07	0.08	0.08	0.01	0.07	(b)
Sept. 14-Sept. 21	0.06	0.04	0.05	0.04	0.05	0.05	0.06	0.04	0.05	0.07	0.04
Sept. 21-Sept. 28	0.05	0.06	0.05	0.05	0.05	0.04	(b)	0.04	0.05	(b)	0.07
Sept. 28-Oct. 5	0.04	0.04	0.02	0.25	0.03	0.03	0.04	0.03	0.04	0.03	0.03
Oct. 5-Oct. 12	0.03	0.03	0.03	0.03	0.04	0.02	0.02	0.03	0.02	0.02	0.03
Oct. 12-Oct. 19	0.04	0.04	0.03	0.03	0.05	0.04	0.04	0.03	0.05	0.04	0.06
Oct. 19-Oct. 26	0.06	0.04	0.03	0.04	0.04	0.04	0.03	0.04	0.04	0.04	0.04
Oct. 26-Nov. 2	0.02	0.04	0.03	0.03	0.03	0.04	0.03	0.02	0.03	0.03	0.04
Nov. 2-Nov. 10	(b)	0.01	0.02	0.03	0.03	0.02	0.02	(b)	0.03	0.01	0.01
Nov. 10-Nov. 16	0.03	0.03	0.02	0.01	0.02	0.03	0.02	0.01	0.02	0.02	0.02

a. Error is ±10% or 0.01, whichever is larger.
b. Instrument malfunction--no data available.

Table 7

Air Particulates--Gross Gamma Concentrations in Monthly Composites (cpm/m^3)[a,b]

Collection Period (1972)	Control	Offsite			Onsite						
	East Weymouth	Plymouth Center	Cleft Rock	Manomet Substation	West Rocky Hill Road	Overlook Area	Pedestrian Bridge	East Rocky Hill Road	Property Line	East Breakwater	Warehouse Sampler
July	24	31	27	28	36	28	22	27	9	24	18
August	17	11	14	17	13	9	13	10	12	10	11
September	13	15	14	22	11	16	14	14	15	11	(c)
October	5	1	5	7	6	5	6	5	4	5	5
November	I[d]	I	I	I	I	I	I	I	I	I	I

a. All values are to be multiplied by 10^{-3}.
b. Error is ±2σ or 10%, whichever is larger.
c. Instrument malfunction--no data available.
d. I= incomplete analysis. Results will be reported in next semiannual report.

Table 8

Gaseous and Particulate Iodine-131 in Air Samples (pCi/m^3)[a]

Collection Period (1972)	Control	Offsite			Onsite						
	East Weymouth	Plymouth Center	Cleft Rock	Manomet Substation	West Rocky Hill Road	Overlook Area	Pedestrian Bridge	East Rocky Hill Road	Property Line	East Breakwater	Warehouse Sampler
July 6-July 13	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
July 13-July 21	"	"	"	"	"	"	"	"	"	"	"
July 21-July 27	"	"	"	"	"	"	"	"	"	"	"
July 27-Aug. 3	"	"	"	"	"	"	"	"	"	"	"
Aug. 3-Aug. 10	"	"	"	"	"	"	"	"	"	"	"
Aug. 10-Aug. 17	"	"	"	"	"	"	"	"	"	"	"
Aug. 17-Aug. 24	"	"	"	"	"	"	"	"	"	"	"
Aug. 24-Aug. 30	"	"	"	"	"	"	"	"	"	"	"
Aug. 30-Sept. 7	"	"	"	(b)	"	"	"	"	"	"	(b)
Sept. 7-Sept. 14	"	"	"	<1	"	"	"	"	"	"	(b)
Sept. 14-Sept. 21	"	"	"	"	"	"	"	"	"	"	<1
Sept. 21-Sept. 28	"	"	"	"	"	"	(b)	"	"	(b)	"
Sept. 28-Oct. 5	"	"	"	"	"	"	<1	"	"	<1	"
Oct. 5-Oct. 12	"	"	"	"	"	"	"	"	"	"	"
Oct. 12-Oct. 19	"	"	"	"	"	"	"	"	"	"	"
Oct. 19-Oct. 26	"	"	"	"	"	"	"	"	"	"	"
Oct. 26-Nov. 2	"	"	"	"	"	"	"	"	"	"	"
Nov. 2-Nov. 10	(b)	"	"	"	"	"	"	"	"	"	"
Nov. 10-Nov. 16	<1	"	"	"	"	"	"	"	"	"	"

a. Both particulate iodine-131 concentrations on filters and gaseous iodine-131 concentrations on impregnated filters were less than minimum detectable concentrations in all samples.
b. Instrument malfunction--no data available.

PILGRIM NUCLEAR GENERATING STATION

Environmental Radiation Monitoring Program

SEMIANNUAL REPORT NO. 2

JANUARY 1, 1973 THROUGH JUNE 30, 1973

DATE OF ISSUE: AUGUST 29, 1973

BOSTON EDISON COMPANY

I. ABSTRACT

The operational environmental radiation surveillance program continues to be conducted in accordance with the specifications in the Pilgrim Station operating license. Reactor operation during the reporting period averaged 85% of full power. Integrated radionuclide discharges in the gaseous effluent increased over the first semi-annual period due to the increased plant capacity factor. Liquid effluents were maintained at low levels by minimum liquid releases and improved radwaste treatment. A detailed post-operational radiation survey was made in January 1973 using a pressurized ion chamber and NaI spectrometer for comparison to a similar preoperational survey conducted in July 1972.

No significant increases in environmental radiation and radioactivity levels beyond the station property were evident in the monitoring data.

Onsite radiation levels increased in areas close to the turbine building. The increases were attributed to direct radiation from the N-16 present in the turbine steam. The largest increase of any consequence (about 12 uR/h above background) occurred at the overlook area, a visitors' area controlled by Boston Edison Company. The average increased dose to the visitor population at this area over natural background is estimated to be 0.08 man-rem during this reporting period. The dose accumulated by all of the visitor population to the Pilgrim Station (22,000 persons) is estimated at 0.1 man-rem during the reporting period. The guard at the public parking area, the two attendants employed there and at the overlook area, and the four groundskeepers each received an estimated 0.6 mrem over the reporting period. The natural background exposure

over this period was about 50 mrem.

The Cs-137 concentration in the seawater in the discharge canal was slightly higher than that of the intake canal although other reactor produced nuclides such as Mn-54, Co-58, 60 and Zn-65 were not detectable. The discharge levels of Cs-137 into the canal has been similarly as low as these other nuclides so that the slightly higher Cs-137 concentration may or may not be due to reactor operation.

The bottom sediment in the discharge canal outfall has been sampled extensively and showed no significant radioactivity, in particular no Co-58, 60 activity, as found in the discharge canal and reported in the previous report.

II. INTRODUCTION

This is the second semiannual report of the Pilgrim Nuclear Power Station operational environmental radioactivity monitoring program, covering the period from January 1, 1973 through June 30, 1973. The monitoring data are reported in accordance with the operational environmental monitoring and reporting requirements of Appendix A to the Pilgrim Facility Operating License DPR-35, Technical Specifications and Bases.

Pilgrim Station began initial start-up operations in June 1972; however, approximately full power operation was not attained until November 1972. During the current reporting period, gross thermal power generation averaged approximately 85 percent of full power operation. The radionuclide emissions in the gaseous effluents increased above those reported during the previous 6-month period due to increased power generation, however, the emissions remained

Table 1

Air Particulates and Gaseous Radioiodine Surveillance Stations

Sampling Location	Distance and Direction from Station
Offsite Stations	
East Weymouth (EW)	23 miles NW
Plymouth Center (PC)	4.5 miles W-WNW
Manomet Substation (MS)	2.5 miles SE
Cleft Rock Area (CR)	0.9 miles S
Onsite Stations	
Rocky Hill Road (ER)	0.8 miles SE
Rocky Hill Road (WR)	0.3 miles W-WNW
Overlook Area (OA)	0.03 miles S W
Onsite Stations Not Required by Operating License	
Property Line (PL)	0.34 miles NW
Pedestrian Bridge (PB)	0.14 miles N
East Breakwater (EB)	0.35 miles ESE
Warehouse (WS)	0.03 miles SSE

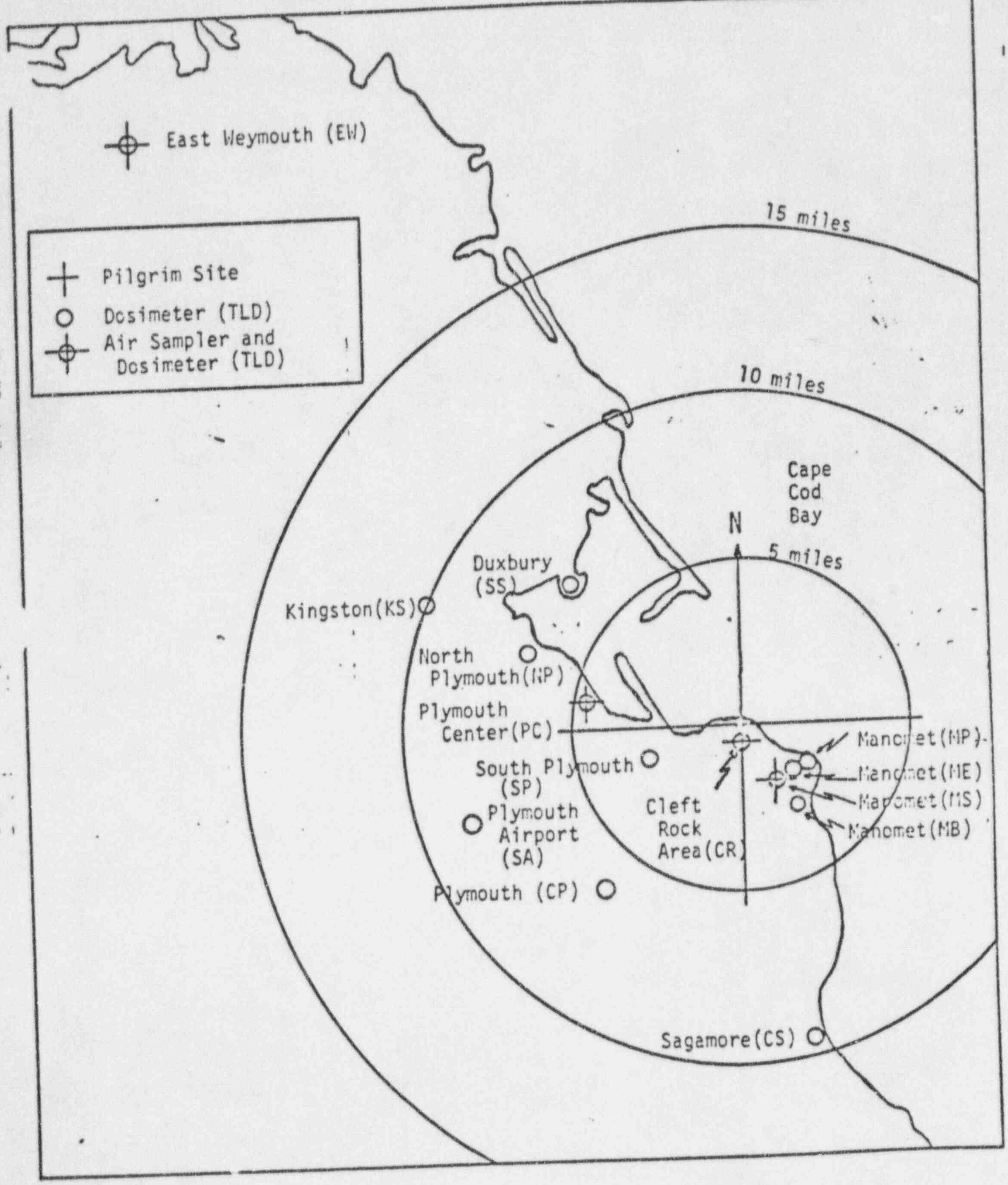

Figure 1. Locations of Offsite Exposure and Air Particulate Monitoring Stations.

Table 6

Air Particulates--Gross Beta Concentrations in Weekly Samples (pCi/m^3)(a)

Collection Period (1973)	Control	Offsite			Onsite						
	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property* Line	Overlook Area	Pedestrian* Bridge	East* Breakwater	Rocky Hill Road (East)	Warehouse*
Jan. 3-Jan. 11	0.03	0.04	0.04	0.03	0.03	0.03	0.03	0.03	0.03	0.03	0.06
Jan. 11-Jan. 18	0.07	0.01	0.06	0.06	0.06	0.05	C.05	0.05	0.05	0.06	0.08
Jan. 18-Jan. 25	0.03	0.03	0.03	0.03	0.03	0.03	0.03	0.02	0.03	0.02	0.03
Jan. 25-Jan. 31	0.03	0.04	0.03	0.03	0.03	0.03	0.03	0.02	0.02	0.03	0.03
Jan. 31-Feb. 8	0.02	0.03	0.02	0.03	0.03	0.02	0.02	0.02	0.02	0.03	0.03
Feb. 8-Feb. 15	0.04	0.02	0.02	0.03	0.02	0.02	0.02	0.02	0.02	0.03	0.02
Feb. 15-Feb. 22	0.04	0.04	0.02	0.04	0.04	0.04	0.03	0.03	0.03	0.02	0.03
Feb. 22-Feb. 28	0.05	0.05	0.03	0.03	0.05	0.06	0.04	0.04	0.05	0.05	0.05
Feb. 28-Mar. 8	0.04	0.04	0.05(b)	0.04	0.03	0.03	0.02	0.04	0.04	0.04	0.04
Mar. 8-Mar. 15	0.04	0.04	0.02	0.04	0.03	0.03	0.03	0.03	(c)	0.04	0.03
Mar. 15-Mar. 22	0.02	0.02	0.01	0.02	0.02	0.02	0.02	0.02	0.02	0.03	0.02
Mar. 22-Mar. 28	0.06	0.05	0.05	0.05	0.05	0.04	0.04	0.05	0.02	0.06	0.05
Mar. 28-Apr. 4	0.02	0.04	0.03	0.03	0.02	0.03	0.02	0.03	(c)	0.04	0.03
Apr. 4-Apr. 12	(c)	0.05	0.04	0.03	0.04	0.04	0.03	0.04	0.04	0.03	0.04
Apr. 12-Apr. 19	(c)	0.05	0.04	0.05	0.05	0.04	0.05	0.07	0.05	0.06	0.05
Apr. 19-Apr. 26	0.04	0.04	0.04	0.03	0.03	0.04	0.02	0.04	0.05	0.04	0.03
Apr. 26-May 3	0.03(d)	0.03	0.03	0.03	0.02	0.03	0.03	0.03	0.03	0.01(d)	0.03
May 3-May 10	0.02	0.01	0.02	0.03	0.02	0.02	0.02	0.03	0.02	0.02	(c)
May 10-May 16	0.04	0.03	0.03	0.03	0.04	0.02	0.02	0.03	0.04	0.03	0.02
May 16-May 24	0.02	0.03	0.02	0.02	0.02	<0.01	0.03	0.02	0.03	0.03	0.02
May 24-May 31	0.02	0.02	0.02	0.02	0.02	0.02	0.01	0.02	0.02	0.02	0.02
May 31-June 7	0.03	0.04	0.03	0.02	0.02	0.01	0.03	0.04	0.03	0.03	0.03
June 7-June 14	0.05	0.06	0.05	0.04	0.04	0.04	0.04	0.05	0.04	0.05	0.04
June 14-June 21	0.02	0.02	0.02	0.01(d)	0.02	0.02	0.01	0.02	0.03	0.03	0.03
June 21-June 28	(e)	0.01	0.03	0.02	0.03	0.02	0.02	0.02	0.02	0.03	0.02
June 28-July 5	0.04	(c)	0.02	0.02	(c)	0.02	0.01(d)	0.02	0.02(d)	0.03	0.02(d)

a. Error is ± 0.01 or 10 percent, whichever is greater.
b. Small volume.
c. No sample.
d. Instrument malfunction--volume estimated.
e. No air particulate filter paper.
* Not required by operating license.

Table 7

Air Particulates--Gross Gamma Concentrations in Monthly Composites (cpm^3 x 10^{-3})(a)

Collection Period (1973)	Control	Offsite			Onsite						
	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property* Line	Overlook Area	Pedestrian* Bridge	East* Breakwater	Rocky Hill Road (East)	Warehouse*
January	7	7	7	5	9	7	5	8	9	10	13
February	2	9	2	7	6	5	6	6	17	12	6
March	5	9	8	5	4	<2	4	4.	7	8	<2
April	4	8	<2	8	5	<2	3	11	<2	6	6
May	<2	<2	<2	<2	<2	<2	<2	<2	<2	<2	<2
June	4	3	3	3	2	<2	<2	3	5	<2	<2

a. Error is ± 2 or 10 percent, whichever is larger.

* Not required by operating license.

Table 8

Gaseous and Particulate Iodine-131 in Air Samples (pCi/m^3)[(a)]

Collection Period (1973)	Control	Offsite			Onsite						
	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property* Line	Overlook Area	Pedestrian* Bridge	East* Breakwater	Rocky Hill Road (East)	Warehouse*
Jan. 3-Jan. 11	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Jan. 11-Jan. 18	"	"	"	"	"	"	"	"	"	"	"
Jan. 18-Jan. 25	"	"	"	"	"	"	"	"	"	"	"
Jan. 25-Jan. 31	"	"	"	"	"	"	"	"	"	"	"
Jan. 31-Feb. 8	"	"	"	"	"	"	"	"	"	"	"
Feb. 8-Feb. 15	"	"	"	"	"	"	"	"	"	"	"
Feb. 15-Feb. 22	"	"	"	"	"	"	"	"	"	"	"
Feb. 22-Feb. 28	"	"	"	"	"	"	"	"	"	"	"
Feb. 28-Mar. 8	"	"	"	"	"	"	"	"	"	"	"
Mar. 8-Mar. 15	"	"	"	"	"	"	"	"	(b)	"	"
Mar. 15-Mar. 22	"	"	"	"	"	"	"	"	<1	"	"
Mar. 22-Mar. 28	"	"	"	"	"	"	"	"	"	"	"
Mar. 28-Apr. 5	"	"	"	"	"	"	"	"	(b)	"	"
Apr. 5-Apr. 12	(b)	"	"	"	"	"	"	"	<1	"	"
Apr. 12-Apr. 19	(b)	"	"	"	"	"	"	"	"	"	"
Apr. 19-Apr. 26	<1	"	"	"	"	"	"	"	"	"	"
Apr. 26-May 3	"	"	"	"	"	"	"	"	"	"	"
May 3-May 10	"	"	"	"	"	"	"	"	"	"	(b)
May 10-May 16	"	"	"	"	"	"	"	"	"	"	<1
May 16-May 24	"	"	"	"	"	"	"	"	"	"	"
May 24-May 31	"	"	"	"	"	"	"	"	"	"	"
May 31-June 7	"	"	"	"	"	"	"	"	"	"	"
June 7-June 14	"	"	"	"	"	"	"	"	"	"	"
June 14-June 21	"	"	"	"	"	"	"	"	"	"	"
June 21-June 28	"	"	"	"	"	"	"	"	"	"	"

a. I-131 concentrations were less than the detection limit in all samples.
b. No sample.
* Not required by operating license.

1972
Table A-6

Air Particulates--Gross Beta Concentrations in Weekly Samples (pCi/m^3)[a]

Collection Period (1972)	Control	Offsite			Onsite		
	East Weymouth	Plymouth Center	Manomet Substation	Cleft Rock	Rocky Hill Road (West)	Rocky Hill Road (East)	Overlook Area
July 6-July 13	0.23	0.26	0.18	0.18	0.22	0.18	0.16
July 13-July 21	0.06	0.04	0.04	0.04	0.04	0.04	0.04
July 21-July 27	0.11	0.14	0.12	0.13	0.13	0.11	0.10
July 27-Aug. 3	0.11	0.11	0.11	0.10	0.11	0.11	0.13
Aug. 3-Aug. 10	0.11	0.09	0.09	0.09	0.09	0.10	0.12
Aug. 10-Aug. 17	0.10	0.09	0.08	0.09	0.07	0.10	0.08
Aug. 17-Aug. 24	0.10	0.09	0.14	0.08	0.08	0.11	0.07
Aug. 24-Aug. 30	0.07	0.06	0.03	0.13	0.04	0.02	0.05
Aug. 30-Sept. 7	0.07	0.07	(b)	0.06	0.07	0.08	0.05
Sept. 7-Sept. 14	0.08	0.07	0.08	0.08	0.07	0.08	0.07
Sept. 14-Sept. 21	0.06	0.04	0.04	0.05	0.05	0.04	0.05
Sept. 21-Sept. 28	0.05	0.06	0.05	0.05	0.05	0.04	0.04
Sept. 28-Oct. 5	0.04	0.04	0.25	0.02	0.03	0.03	0.03
Oct. 5-Oct. 12	0.03	0.03	0.03	0.03	0.04	0.03	0.02
Oct. 12-Oct. 19	0.04	0.04	0.03	0.03	0.05	0.03	0.04
Oct. 19-Oct. 26	0.06	0.04	0.04	0.03	0.04	0.04	0.04
Oct. 26-Nov. 2	0.02	0.04	0.03	0.03	0.03	0.02	0.04
Nov. 2-Nov. 10	(b)	0.01	0.03	0.02	0.03	(b)	0.02
Nov. 10-Nov. 16	0.03	0.03	0.01	0.02	0.02	0.01	0.03
Nov. 16-Nov. 22	0.03	0.04	0.02	0.04	0.03	0.05	0.04
Nov. 22-Nov. 30	0.04	0.04	0.04	0.03	0.03	0.03	0.03
Nov. 30-Dec. 7	0.03	0.03	0.02	0.03	0.03	0.02	0.03
Dec. 7-Dec. 15	0.05	0.05	0.04	0.04	0.03	0.03	(b)
Dec. 15-Dec. 21	0.04	0.04	0.03	0.05	(b)	0.04	0.08
Dec. 21-Dec. 28	0.02	0.03	0.02	0.02	(b)	0.02	0.03
Dec. 28-Jan. 4	0.03	0.03	0.03	0.04	0.04	0.03	0.03

a. Error is ±10 percent or 0.01, whichever is larger.
b. Instrument malfunction--no data available.

1972
Table A-7

Air Particulates--Gross Gamma Concentrations in Monthly Composites (cpm/m^3)[a,b]

Collection Period (1972)	Control: East Weymouth	Offsite: Plymouth Center	Offsite: Manomet Substation	Offsite: Cleft Rock	Onsite: Pocky Hill Road (West)	Onsite: Rocky Hill Road (East)	Onsite: Overlook Area
July	24(c)	31(c,d)	28(c,d)	27(c,d)	36(c)	27(c)	28(c,d)
August	17	11	17	14	13	10	9
September	13	15	22	14	11	14	16
October	5	1(e)	7(e)	5	6(e)	5(e)	5(e)
November	2	<2	<2	<2	4	3	5
December	5	8	4	7	2	7	4

a. All values are to be multiplied by 10^{-3}.
b. Error is $\pm 2\sigma$ or 10 percent.
c. Ru-103, Ru-106, Zr-95, and Nb-95 detected by gamma spectrum analysis.
d. Ce-141 and Ce-144 detected by gamma spectrum analysis.
e. Ru-103 and Ru-106 detected by gamma spectrum analysis.

Table A-8

Gaseous and Particulate Iodine-131 in Air Samples (pCi/m^3)[(a)]

Collection Period (1972)	Control	Offsite			Onsite		
	East Weymouth	Plymouth Center	Manomet Substation	Cleft Rock	Rocky Hill Road (West)	Rocky Hill Road (East)	Overlook Area
July 6-July 13	<1	<1	<1	<1	<1	<1	<1
July 13-July 21	"	"	"	"	"	"	"
July 21-July 27	"	"	"	"	"	"	"
July 27-Aug. 3	"	"	"	"	"	"	"
Aug. 3-Aug. 10	"	"	"	"	"	"	"
Aug. 10-Aug. 17	"	"	"	"	"	"	"
Aug. 17-Aug. 24	"	"	"	"	"	"	"
Aug. 24-Sept. 7	"	"	(b)	"	"	"	"
Sept. 7-Sept. 14	"	"	<1	"	"	"	"
Sept. 14-Sept. 21	"	"	"	"	"	"	"
Sept. 21-Sept. 28	"	"	"	"	"	"	"
Sept. 28-Oct. 5	"	"	"	"	"	"	"
Oct. 5-Oct. 12	"	"	"	"	"	"	"
Oct. 12-Oct. 19	"	"	"	"	"	"	"
Oct. 19-Oct. 26	"	"	"	"	"	"	"
Oct. 26-Nov. 2	"	"	"	"	"	"	"
Nov. 2-Nov. 10	(b)	"	"	"	"	"	"
Nov. 10-Nov. 16	<1	"	"	"	"	"	"
Nov. 16-Nov. 22	"	"	"	"	"	"	"
Nov. 22-Nov. 30	"	"	"	"	"	"	"
Nov. 30-Dec. 7	"	"	"	"	"	"	"
Dec. 7-Dec. 15	"	"	"	"	"	"	(b)
Dec. 15-Dec. 21	"	"	"	"	(b)	"	<1
Dec. 21-Dec. 228	"	"	"	"	(b)	"	"
Dec. 28-Jan. 4	"	"	"	"	<1	"	"

a. Both particulate iodine-131 concentrations on filters and gaseous iodine-131 concentrations on impregnated filters were less than minimum detectable concentrations in all samples.

b. Instrument malfunction--no data available.

TABLE D-2
SEMI-ANNUAL SUMMARY OF RADIOACTIVE GASEOUS EFFLUENTS
JANUARY - JUNE, 1973

		JAN.	FEB.	MAR.	APRIL	MAY	JUNE
Total Noble Gases	(Curies)						
(a) Main Stack		1.65 E4	1.10 E4	1.52 E4	2.02 E4	2.84 E4	1.96 E4
(b) Reactor Building Vent		1.49 E2	1.37 E2	1.47 E2	7.30 E1	5.39 E2	3.08 E2
Total Halongens (*-1)	(Curies)						
(a) Main Stack		1.02 E-2	9.89 E-3	1.46 E-2	1.00 E-2	1.43 E-1	2.01 E-2
(b) Reactor Building Vent		1.23 E-3	9.72 E-4	2.09 E-3	1.24 E-2	4.18 E-2	5.32 E-3
Total Particulate Gross Beta-Gamma Radioactivity (*-1)	(Curies)						
(a) Main Stack		1.20 E-4	8.26 E-5	8.98 E-4	1.07 E-3	8.12 E-4	1.31 E-3
(b) Reactor Building Vent		1.19 E-4	1.01 E-4	2.68 E-4	6.57 E-4	1.72 E-4	1.63 E-4
Total Particulate Gross Alpha Radioactivity	(Curies)	(*-2)	(*-2)	(*-2)	(*-2)	(*-2)	(*-2)
Total Tritium	(Curies)						
(a) Main Stack		1.25 E-1	1.52 E-1	2.19 E-1	1.37 E-1	2.14 E-1	1.94 E-1
(b) Reactor Building Vent		1.39 E0	8.20 E-2	1.07 E0	1.81 E0	1.67 E0	1.26 E0
Maximum 24-Hr. Noble Gas Release	(Curies)	1-17 7.67 E2	2-8 6.48 E2	3-12 8.45 E2	4-30 1.03 E3	5-27 1.56 E3	6-29 1.09 E3
Percent of Applicable Limit for Noble Gases		2.50	1.87	2.33	3.15	4.44	3.14
Percent of Applicable Limit for Halogens & Particulates		1.15	1.06	1.45	8.73	30.07	4.14
Isotopes Released	(Curies)						
A. Halogens							
Iodine-131		1.14 E-2	1.09 E-2	1.67 E-2	2.24 E-2	1.85 E-1	2.54 E-2
Iodine-133		(*-3)	(*-3)	3.92 E-2	(*-3)	5.41 E-2	(*-3)
Iodine-135		(*-3)	(*-3)	(NDA*-4)	(*-3)	(NDA*-4)	(*-3)
B. Particulates							
Barium/Lanthanum-140		2.21 E-5	3.51 E-5	4.66 E-4	3.10 E-4	3.87 E-4	4.79 E-4
Beryllium-7					1.53 E-5	5.42 E-6	
Cesium-134					5.67 E-7	1.87 E-6	
Cesium-137		4.26 E-7	8.20 E-6	7.49 E-6	8.56 E-6	1.63 E-5	8.16 E-6
Chromium-51		1.49 E-5	1.73 E-4	3.07 E-4	9.62 E-5	2.97 E-4	1.49 E-4
Cobalt-58		3.49 E-6	3.82 E-5	3.78 E-5	4.72 E-6	2.65 E-5	1.18 E-4
Cobalt-60				2.99 E-6	1.99 E-6	1.06 E-5	3.22 E-6
Iron-59						3.56 E-6	1.89 E-6
Manganese-54					2.32 E-6	1.96 E-5	6.21 E-6
Zinc-65						1.91 E-6	1.30 E-6
Zirconium-95						4.52 E-7	
Niobium-95						1.82 E-7	
C. Gases (*-5)							
Xenon-138		4.27 E2	3.84 E2	4.73 E2	6.48 E2	8.12 E2	6.25 E2
Krypton-87		1.72 E3	1.57 E3	1.88 E3	3.25 E3	4.17 E3	2.39 E3
Krypton-88		2.48 E3	1.77 E3	2.26 E3	3.57 E3	4.61 E3	2.96 E3
Krypton-85m		1.05 E3	7.34 E2	1.19 E3	1.56 E3	2.01 E3	1.37 E3
Xenon-135		4.09 E3	2.80 E3	3.88 E3	6.12 E3	8.81 E3	5.43 E3
Xenon-133		2.72 E3	2.63 E3	4.16 E3	4.29 E3	6.34 E3	5.03 E3
Sum of Remainder		4.02 E3	1.10 E3	1.36 E3	7.62 E2	1.64 E3	1.81 E3

(*-1) With half-lives greater than 8 days
(*-2) Not measured since no alpha found in reactor coolant
(*-3) Quarterly analysis - Tech. Spec.
(*-4) NDA = No detectable activity
(*-5) Main stack only

GJD

PILGRIM NUCLEAR GENERATING STATION

Environmental Radiation Monitoring Program

SEMIANNUAL REPORT NO. 3

JULY 1, 1973 THROUGH DECEMBER 31, 1973

DATE OF ISSUE: MARCH 1, 1974

BOSTON EDISON COMPANY

ABSTRACT

The operational environmental radiation surveillance program continues to be conducted in accordance with the specifications in the Pilgrim Station operating license. Reactor operation during the reporting period averaged about 60% of full power. An administrative limit of 50% of full power was maintained from October 6 until the maintenance and refueling shutdown on December 28. Effluent discharges decreased as compared to the previous semiannual period, primarily as a result of reduced plant capacity factor. Detailed operating information, including a summary of plant releases, is reported in a separate semiannual report entitled, "Operating and Maintenance Report."

No significant increases in environmental radiation or radioactivity levels beyond the station property were evident in the monitoring data.

Radiation levels in areas close to the turbine building continued to show an increase over background. The increase was attributed to direct radiation from the N-16 present in the turbine steam. Two onsite visitor areas were subjected to this increase. As a result, the increased radiation dose to the 57,000 visitors over the reporting period was estimated to be 0.25 man-rem. Increased exposures were also calculated for transient workers, working outside of the security area. Two visitor area attendants each received about 1.7 millirem (mrem), while a third received about 0.9 mrem. Four groundskeepers absorbed approximately one millirem each and two research workers each received less than 0.4 mrem of increased radiation dose, over the reporting period. The background dose over this period was about 50 mrem.

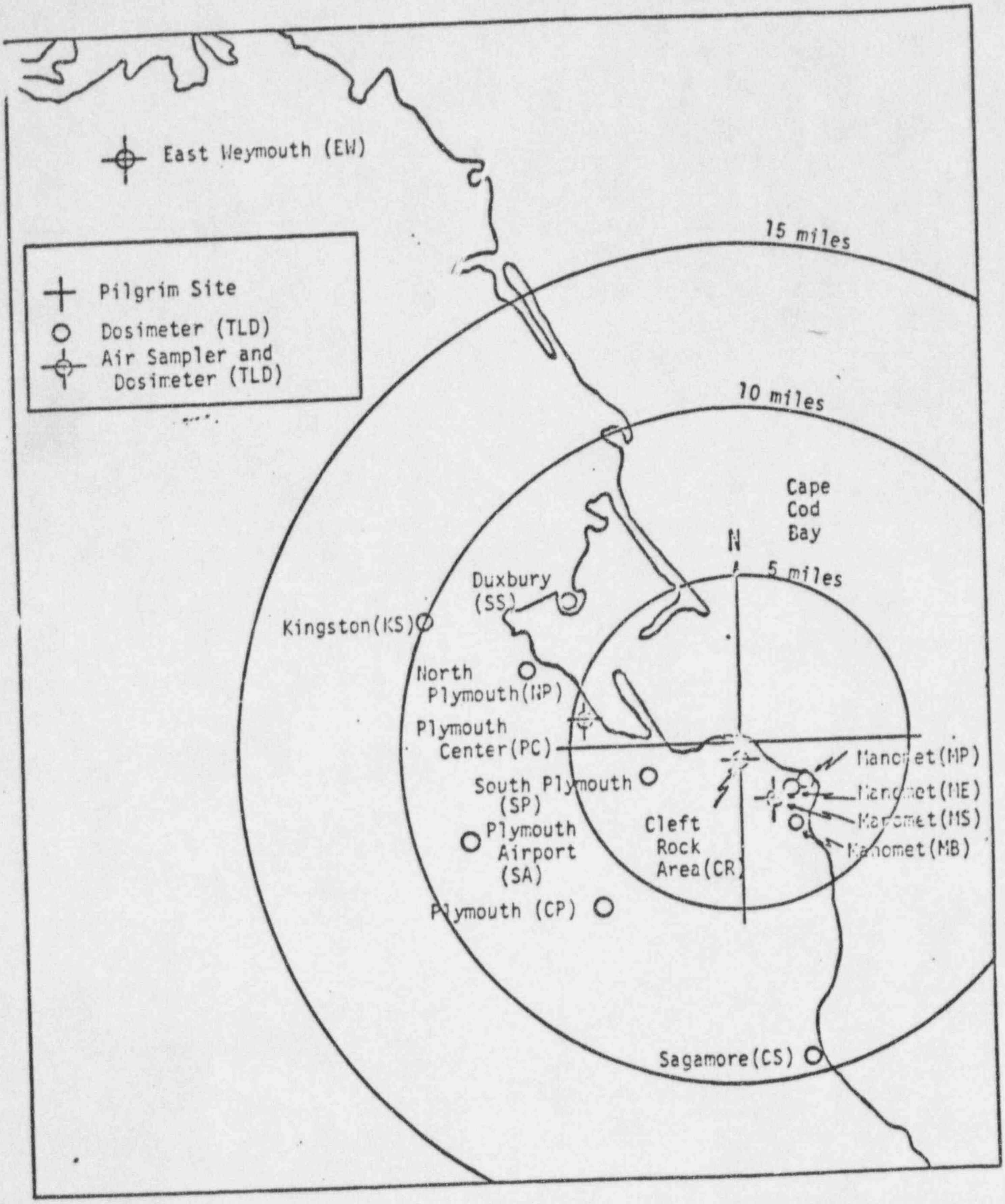

Figure 1. Locations of Offsite Exposure and Air Particulate Monitoring Stations.

Table 6

Air Particulates--Gross Beta Concentrations in Weekly Samples (pCi/m^3)(a)

Collection Period (1973)		Control	Offsite				Onsite					
		East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property* Line	Overlook Area	Pedestrian* Bridge	East* Breakwater*	Rocky Hill Road (East)	Warehouse*
June 28-July	5	0.04	(c)	0.02	0.02	(c)	0.02	0.01(d)	0.02	0.02(d)	0.03	0.02(d)
July 5-July	12	0.06	0.01	0.07	0.06	(c)	0.07	0.01(d)	0.07	0.05	0.09	0.06
July 12-July	19	0.04	0.06(b)	0.04	0.03	0.03	0.05	0.04	0.05	0.04	0.04	0.02
July 19-July	26	0.06	0.06	0.04	0.04	0.05	0.05	0.06	0.05	0.05	0.04	[illegible]
July 26-August	2	0.04	0.04	0.03	0.03	0.03	0.03	0.04	0.03	0.03	0.03	0.03
August 2-August	9	0.05	0.05	0.04	0.04	0.05	0.04	0.05	0.04	0.04	0.04	0.04
August 9-August	16	0.04	0.04	<0.01	0.04	0.04	0.03	0.04	0.03	0.04	0.04	0.01
August 16-August	23	0.01(d)	0.03	0.01	0.03	0.03	0.03	0.04	0.02	0.02	0.03	0.03
August 23-August	30	0.14	0.09	0.03	0.07	0.03	0.07	0.08	0.03	0.07	0.06	0.03
August 30-Sept.	6	0.05	0.06	(c)	0.05	0.07	0.05	0.05	0.06	0.07	(c)	0.05
Sept. 6-Sept.	13	0.05	0.06	(c)	(c)	0.04	0.04	0.04	0.04	0.03	0.03	0.02
Sept. 13-Sept.	20	0.04	0.04	(c)	(c)	0.03	0.03	0.03	0.05	0.13(b)	0.03	0.04
Sept. 20-Sept.	27	0.04	0.04	0.02	0.03	0.02	0.03	0.03	0.05	0.02	0.02	0.02
Sept. 27-Oct.	3	<0.01	0.02	0.01	0.01	0.02	<0.01	0.02	0.01(d)	0.02	0.01	<0.01(d)
Oct. 3-Oct.	11	0.03	0.04	0.03	0.03	0.03	(c)	0.01	0.03	0.03	0.03	0.03
Oct. 11-Oct.	18	0.03	0.04	0.03	0.04	0.03	0.04	0.01	0.04	0.04	0.04	0.04
Oct. 18-Oct.	25	0.04	0.05	0.02	0.04	0.04	0.04	0.04	0.05	0.03	0.04	0.04
Oct. 25-Nov.	1	0.02	0.02	0.02	0.03	0.03	<0.01	0.03	0.03	0.04	0.05	0.03
Nov. 1-Nov.	7	0.05	0.02	0.04	0.03	0.07	0.04	0.06	0.04	0.01	0.05	0.04
Nov. 7-Nov.	15	0.03	0.04	<0.01	0.04	0.04	0.04	0.04	0.04	0.01	0.04	0.04
Nov. 15-Nov.	21	0.03	0.03	0.02	0.03	0.03	(a)	0.04	0.04	(a)	(a)	0.03
Nov. 21-Nov.	29	0.03	0.03	0.03	0.04	0.04	0.04	0.04	<0.01	0.03	0.02	0.03
Nov. 29-Dec.	6	0.05	0.05	0.04	0.08	0.06	0.05	0.08	0.05	0.05	0.05	0.05
Dec. 6-Dec.	13	0.06	0.04	0.03	0.04	0.04	0.05	0.05	0.04	0.03	0.04	0.04
Dec. 13-Dec.	20	0.03	0.04	0.02	0.03	0.04	0.03	0.03	0.03	0.04	0.03	0.12(b)
Dec. 20-Dec.	27	0.04	0.06	0.05	0.05	0.06	0.05	0.06	0.04	0.02	0.04	0.05

a. Error is ±0.01 or 10%, whichever is larger.
b. Small volume--instrument malfunction.
c. No sample.
d. Air volume estimated.
* Station not required by operating license.

Table 7

Air Partuculates--Gross Gamma Concentration in Monthly Composites (cpm/m^3 x 10^{-3})(a)

Collection Period (1973)	Control	Offsite				Onsite					
	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property* Line	Overlook Area	Pedestrian* Bridge	East* Breakwater	Rocky Hill Road (East)	Warehouse*
July	4	10(b)	5	8	8(b)	5	1	6	5	4	3
August	10	10	4	8	12	4	11	12	7	11	29
September	<2	<2	18	4	4	4	4	<2	14	11	<2
October	<2	<2	<2	<2	<2	<2	<2	4	4	7	<2
November	4	2	<2	8	9	4	8	3	16	3	6
December	4	3	3	7	7	10	7	3	3	3	4

a. Error is ±2 or 10%, whichever is larger.
b. Value approximate--air volume estimated.
* Station not required by operating license.

8

Gaseous and Particulate Iodine-131 in Air Samples (pCi/m^3)[a]

Collection Period (1973)	Control East Weymouth	Offsite Plymouth Center	Offsite Cleft Rock Area	Offsite Manomet Substation	Onsite Rocky Hill Road (West)	Onsite Property* Line	Onsite Overlook Area	Onsite Pedestrian* Bridge	Onsite East* Breakwater	Onsite Rocky Hill Road (East)	Onsite Warehouse*
June 22-July 5	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
July 5-July 12	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
July 12-July 19	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
July 19-July 26	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
July 26-August 2	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
August 2-August 9	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
August 9-August 16	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
August 16-August 23	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
August 30-Sept. 6	<1	<1	(b)	<1	<1	<1	<1	<1	<1	<1	<1
Sept. 6-Sept. 13	<1	<1	(b)	(b)	<1	<1	<1	<1	<1	<1	<1
Sept. 13-Sept. 20	<1	<1	(b)	(b)	<1	<1	<1	<1	<1	<1	<1
Sept. 20-Sept. 27	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Sept. 27-Oct. 3	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Oct. 3-Oct. 11	<1	<1	<1	<1	<1	(b)	<1	<1	<1	<1	<1
Oct. 11-Oct. 18	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Nov. 1-Nov. 7	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Nov. 7-Nov. 15	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Nov. 15-Nov. 21	<1	<1	<1	<1	<1	(b)	<1	<1	(b)	(b)	<1
Nov. 21-Nov. 29	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Nov. 29-Dec. 6	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Dec. 6-Dec. 13	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Dec. 13-Dec. 20	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Dec. 20-Dec. 27	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1

a. I-131 concentrations were less than detection limit in all samples.
b. No sample.
* Station not required by operating license.

PILGRIM NUCLEAR GENERATING STATION

Environmental Radiation Monitoring Program

SEMIANNUAL REPORT NO. 4
JANUARY 1, 1974 THROUGH JUNE 30, 1974

ISSUED:
AUGUST 29, 1974

BY:
NUCLEAR ENGINEERING DEPT.
ENVIRONMENTAL SCIENCES GROUP

BOSTON EDISON COMPANY

I. INTRODUCTION AND SUMMARY

The Operational Environmental Radiation Surveillance Program continues to be conducted in accordance with the specifications of the Pilgrim Station Operating License. The reactor was shut down during the entire reporting period for refueling and maintenance and for contested licensing hearings regarding a change in fuel design. As a result, gaseous effluent releases were very low and direct radiation from the facility was exceedingly small. The gaseous Iodine -131 release for January, highest of the semi-annual period, was only about 1.5 percent of the release limit.*

Due to the increased in-plant maintenance activity, the liquid effluent releases were higher than during normal operation.** This fact, coupled with intermittent operation of the circulating (dilution) water pumps, resulted in radionuclide buildup in the discharge canal. The highest level of 5.2 pCi Co-60 per gram sediment was detected in the clay found on the rocks at the sides of the canal. The Co-60 activity level decreased to 0.14 pCi/gm adjacent to the end of the canal and 0.05 pCi/gm 400 feet north-west of the canal. No activity was detected southeast of the station (MDA = 0.01 pCi/gm).

The other media samples showed only radioactivity from natural sources and fallout from the Chinese nuclear weapon test of December 1973. No plant-related nuclides were detected in these media.

*Radioactive effluent releases are summarized in the Appendix.

**Details of plant operation can be found in a separate report entitled, "Operating and Maintenance Semiannual Report #4."

II. DESCRIPTION OF MONITORING PROGRAM

The Environmental Radiation and Radioactivity Surveillance Program is being performed in accordance with the requirements specified in the Pilgrim Facility Operating License (DPR-35). Summaries of the sampling media, locations, frequencies of collection and analyses are given in Tables 1 through 4. Details of the sampling program are given in Semiannual Report #2.* Sampling locations are shown in Figures 1, 2, and 3.

The radio-analysis of environmental samples is being performed by Interex Corporation. The limits of detection associated with their analytical procedures are given in Table 5. Samples of bottom sediment and selected marine life were analyzed by Teledyne Isotopes using Ge(Li) gamma spectrometry. The sensitivities of their analyses are given with the results in the appropriate tables.

*Pilgrim Nuclear Generating Station, Environmental Radiation Monitoring Program, Semiannual Report #2, August 29, 1973.

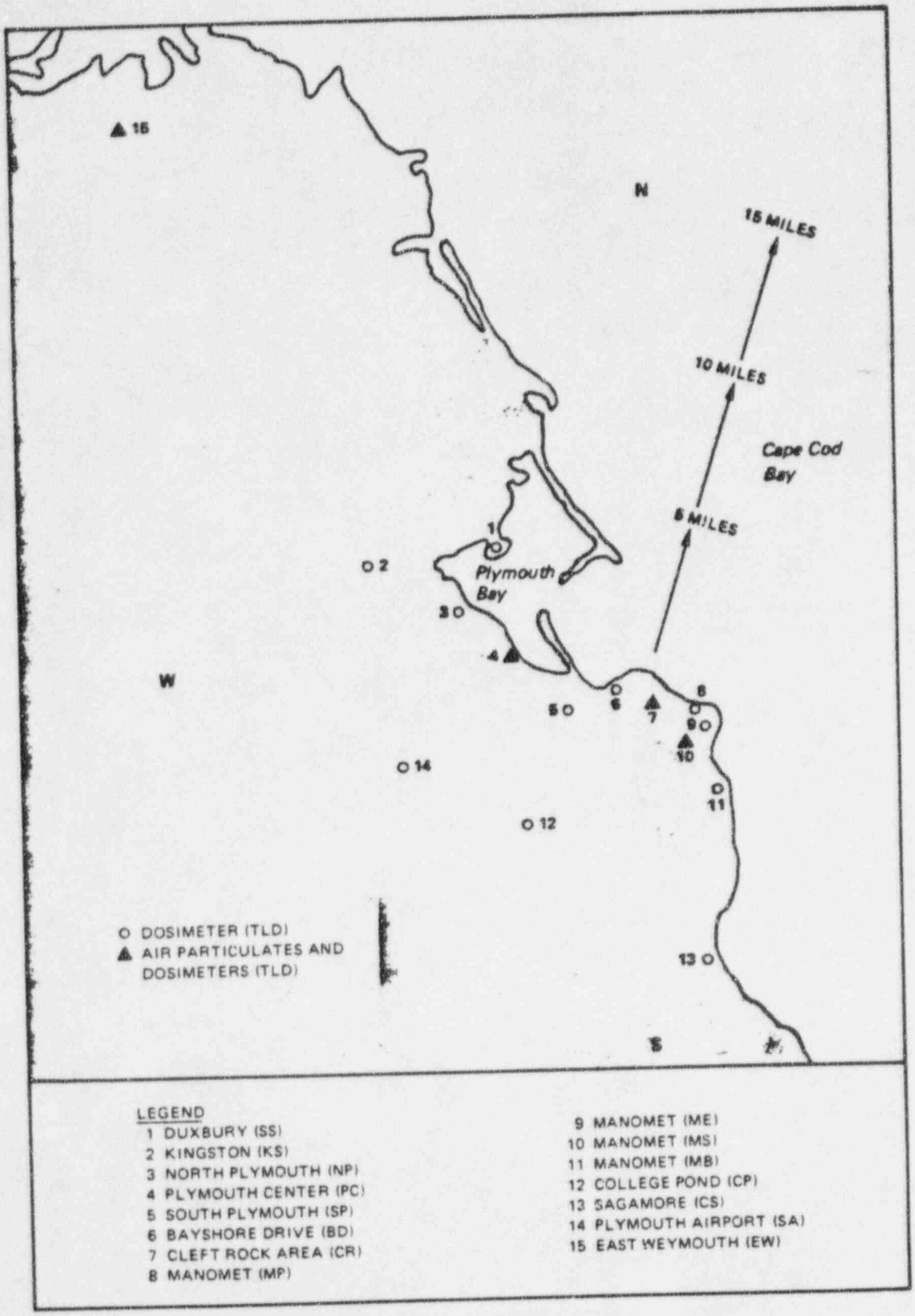

Figure 1. Location of Offsite Radiological Monitoring Stations

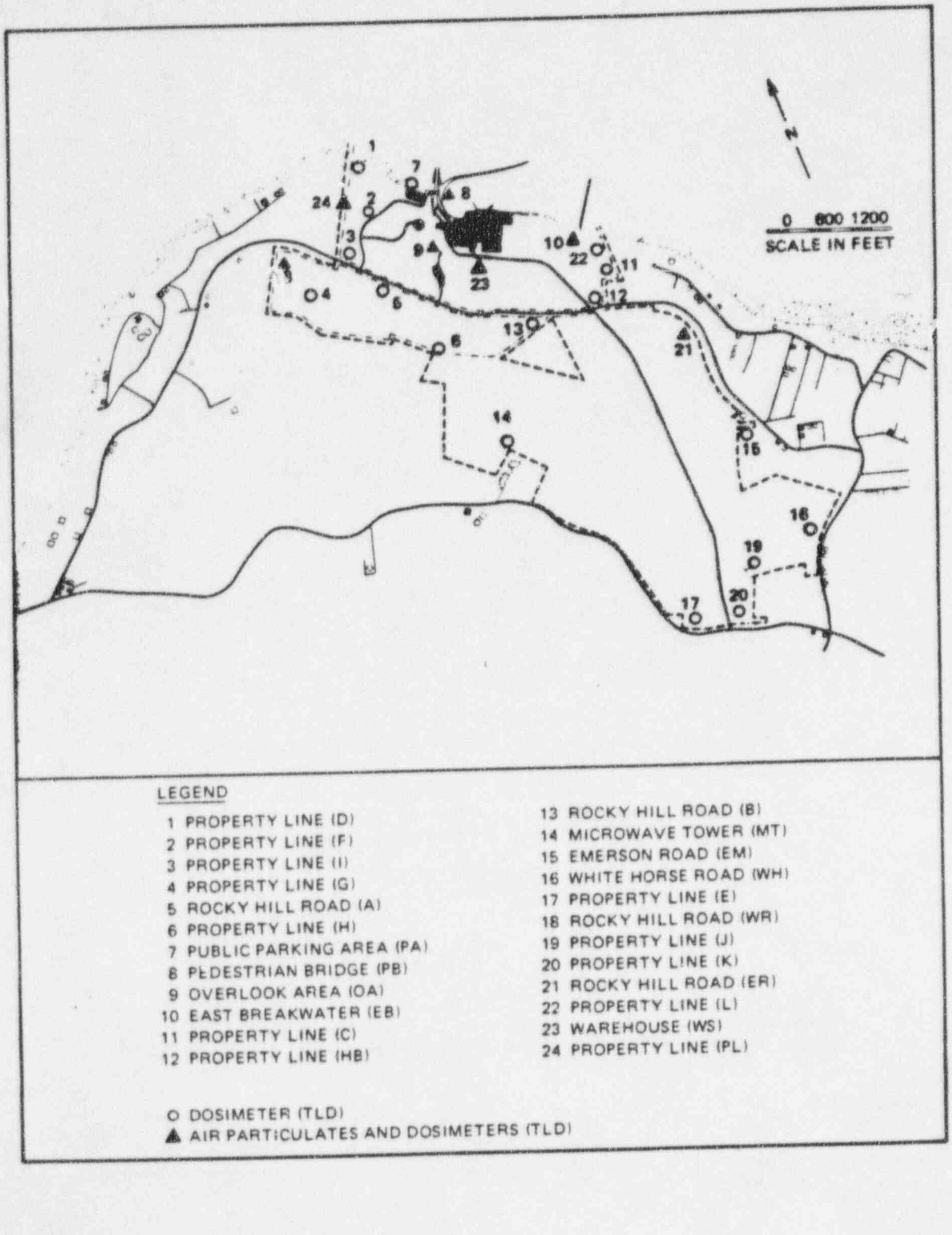

Figure 2. Location of Onsite Radiological Monitoring Stations

TABLE 6

AIR PARTICULATES - GROSS BETA CONCENTRATIONS IN WEEKLY SAMPLES (pCi/m^3)

Collection Period (1974)	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property Line*	Overlook Area	Pedestrian Bridge*	East Breakwater*	Rocky Hill Road (East)	Warehouse*
Dec 27 - Jan 3	0.03	0.04	0.04	0.04	0.04	0.05	0.05	0.04	0.04	0.04	0.04
Jan 3 - Jan 10	(a)	0.04	(a)	(a)	0.04	0.05	(a)	0.07	(a)	0.04	0.04
Jan 10 - Jan 17	0.02	0.05	0.04	0.02	0.04	0.04	0.10	0.12	0.04	0.06	0.04
Jan 17 - Jan 24	0.09	0.05	0.08	0.05	0.05	0.06	0.05	0.05	0.07	0.05	0.06
Jan 24 - Jan 31	0.08	0.07	0.07	0.08	0.09	0.08	0.10	0.08	0.05	0.08	0.08
Jan 31 - Feb 7	0.07	0.07	0.07	0.07	0.08	0.08	0.07	0.07	0.08	0.07	0.08
Feb 7 - Feb 14	0.07	0.08	0.08	0.07	0.09	0.08	0.08	0.07	0.07	0.07	0.08
Feb 14 - Feb 21	0.07	0.07	0.07	0.08	0.09	0.09	0.10	0.06	0.09	0.09	0.10
Feb 21 - Feb 28	0.13	0.12	0.12	0.12	0.13	0.12	0.15	0.07	0.15	0.12	(c)
Feb 28 - Mar 7	0.16	0.15	0.16	0.15	0.16	0.16	0.18	0.17	0.19	0.16	0.18
Mar 7 - Mar 14	0.14	0.15	0.13	0.13	0.15	0.16	0.15	0.13	0.15	0.13	0.14
Mar 14 - Mar 21	0.06	0.14	0.06	0.15	0.15	0.15	0.06	0.06	0.13	0.11	0.18
Mar 21 - Mar 28	0.18	0.18	0.16	0.15	0.16	0.19	0.20	0.16	0.15	0.15	0.18
Mar 28 - Apr 4	0.11	0.11	0.11	0.10	0.12	0.13	0.13	0.12	0.12	0.10	0.09
Apr 4 - Apr 11	0.23	0.21	0.16	0.17	0.16	0.17	0.20	0.19	0.20	0.21	(c)
Apr 11 - Apr 18	0.22	0.21	0.18	0.20	0.20	0.22	0.22	0.22	0.20	0.22	(c)
Apr 18 - Apr 25	0.26	0.36	0.16	0.31	6.35	0.33	0.37	0.33	0.29	0.31	(c)
Apr 25 - May 2	0.33	0.31	0.38	0.33	0.33	0.38	0.33	0.34	0.29	0.35	0.32
May 2 - May 9	0.28	0.27	0.26	0.26	0.27	0.26	0.28	0.26	0.26	0.28	0.29
May 9 - May 16	0.23	0.20	0.19	0.15	0.20	0.22	0.21	0.20	0.23	0.09 (b)	0.20
May 16 - May 23	0.10	0.29	0.31	0.30	0.32	0.28	0.31 (b)	0.30	0.27	0.28	0.31
May 23 - May 30	0.08	0.10	0.08	0.10	0.09	0.09	(c)	0.11	0.10	0.07	0.11
May 30 - June 6	0.30	0.28	0.29	0.26	0.31	0.28	0.28	0.29	0.29	0.29	0.30
June 6 - June 13	0.24	0.22	0.21	0.23	0.33	0.23	0.22	0.25	0.23	0.22	0.22
June 13 - June 20	0.23	0.15	0.18	0.20	0.21	0.21	0.15	0.21	0.21	0.12	0.15
June 20 - June 27	0.21	0.20	0.14	0.19	0.19	0.19	0.09	0.16	(c)	0.15	0.16

(a) No sample - station snowed in.
(b) Estimated volume - Instrument Malfunction.
(c) No sample.
* Station not required by operating license.

TABLE 7

AIR PARTICULATES - GROSS GAMMA CONCENTRATION IN MONTHLY COMPOSITES ($cpm/m3 \times 10^{-3}$) [a]

Collection Period (1974)	Control	Offsite			Onsite						
	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property Line	Overlook Area	Pedestrian Bridge	East Breakwater	Rocky Hill Road (East)	Warehouse
Jan	6 [b]	11	7	10	9	15	25	12	14	11	12
Feb	14 [b]	14	18	18	14	18	14	4	18	18	14
Mar	15	11	12	12	18	16	13	12	21	I [c]	16
Apr	11	I	I	19	I	26	23	22	24	21	12

(a) Error is ± 2 or 10%, whichever is larger.
(b) Gamma spectrum indicates no nuclides identified greater than 0.05 pCi/m^3.
(c) I indicates incomplete analysis; results will be reported in next semiannual report.

TABLE 8

PARTICULATE IODINE-131 IN AIR SAMPLES (pCi/m^3)

	Control	Offsite			Onsite						
Collection Period	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road West Onsite	Property Line*	Overlook Area	Pedestrian Bridge*	East Breakwater*	Rocky Hill Road East Onsite	Warehouse*
Jan 3 - Jan 10	(a)	<1	(a)	(a)	<1	<1	(a)	<1	(a)	<1	<1
Jan 10 - Jan 17	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Jan 17 - Jan 24	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Jan 24 - Jan 31	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Jan 31 - Feb 7	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Feb 7 - Feb 14	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Feb 14 - Feb 21	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Feb 21 - Feb 28	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Feb 28 - Mar 7	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Mar 7 - Mar 14	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Mar 14 - Mar 21	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Mar 21 - Mar 28	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Mar 28 - Apr 4	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Apr 4 - Apr 11	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	(a)
Apr 11 - Apr 18	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	(a)
Apr 18 - Apr 25	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	(a)
Apr 25 - May 2	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
May 2 - May 9	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
May 9 - May 16	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
May 16 - May 23	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
May 23 - May 30	<1	<1	<1	<1	<1	<1	(a)	<1	<1	<1	<1
May 30 - June 6	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
June 6 - June 13	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1

(a) No sample.
* Station not required by operating license.

TABLE 8A

GASEOUS IODINE-131 IN AIR SAMPLES (pCi/m^3)

	Control	Offsite			Onsite						
Collection Period	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road West	Property Line	Overlook Area	Pedestrian Bridge	East Breakwater*	Rocky Hill Road East Onsite	Warehouse*
Jan 3 - Jan 10	(a)	<1	(a)	(a)	<1	<1	(a)	<1	(a)	<1	<1
Jan 10 - Jan 17	(a)	<1	(a)	(a)	<1	<1	<1	<1	<1	<1	<1
Jan 17 - Jan 24	(a)	(a)	(a)	(a)	(a)	(a)	(a)	(a)	(a)	(a)	(a)
Jan 24 - Jan 31	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Jan 31 - Feb 7	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Feb 7 - Feb 14	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Feb 14 - Feb 21	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Feb 21 - Feb 28	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Feb 28 - Mar 7	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Mar 7 - Mar 14	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Mar 14 - Mar 21	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Mar 21 - Mar 28	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Mar 28 - Apr 4	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Apr 4 - Apr 11	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	(a)
Apr 11 - Apr 18	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	(a)
Apr 18 - Apr 25	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	(a)
Apr 25 - May 2	(b)	(b)	(b)	(b)	(b)	(b)	(b)	(b)	(b)	(b)	(b)
May 2 - May 9	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
May 9 - May 16	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
May 16 - May 23	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
May 23 - May 30	<1	<1	<1	<1	<1	<1	(a)	<1	<1	<1	<1
May 30 - June 7	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
June 7 - June 13	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
June 13 - June 20	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1

(a) No sample.
(b) Sample lost during laboratory move.
* Station not required by operating license.

TABLE A-1

SEMIANNUAL SUMMARY OF RADIOACTIVE LIQUID EFFLUENTS
JANUARY-JUNE, 1974

		Jan	Feb	Mar	Apr	May	June	Total
1. Gross Radioactivity								
(a) Total Released	(Ci)	6.78E-1	1.08E0	7.54E-1	5.14E-1	2.76E-1	4.83E-1	3.79E0
(b) Avg Conc Released	(μCi/ml)	9.50E-8	3.12E-7	3.43E-7	3.86E-7	2.71E-7	1.52E-7	2.07E-7
(c) Max Conc Released	(μCi/ml)	2.03E-7	2.28E-6	1.17E-6	1.10E-6	1.09E-6	1.00E-6	2.28E-6
2. Tritium								
(a) Total Released	(Ci)	3.53E0	2.59E0	1.90E0	1.01E0	5.46E-1	7.22E-1	1.03E1
(b) Avg Conc Released	(Ci/ml)	4.94E-7	7.55E-7	8.64E-7	7.59E-7	5.35E-7	2.28E-7	[illegible]
3. Dissolved Noble Gases								
(a) Total Released	(Ci)	*	*	*	*	*	*	*
(b) Avg Conc Released	(Ci/ml)							
4. Gross Alpha Radioactivity								
(a) Total Released	(Ci)	5.80E-5	2.75E-4	3.00E-4	2.32E-4	5.92E-4	8.77E-3	1.02E-2
(b) Avg Conc Released	(Ci/ml)	3.17E-12	1.50E-11	1.64E-11	1.27E-11	5.80E-10	2.77E-9	5.57E-10
5. Volume of Liquid Waste to Discharge Canal	(Liters)	2.52E6	1.62E6	1.58E6	1.01E6	6.5E5	1.11E6	8.50E6
6. Volume of Dilution Water	(Liters)	7.14E9	3.43E9	2.20E9	1.33E9	1.02E9	3.17E9	1.83E10
7. Isotopes Released	(Ci)							
Chromium-51		3.02E-2	6.51E-3	5.21E-2	7.27E-3	1.79E-3		9.79E-2
Manganese-54		3.53E-2	4.70E-2	4.60E-2	5.35E-2	4.04E-2	2.97E-2	2.52E-1
Cobalt-58		4.28E-2	4.86E-2	1.17E-1	3.74E-2	2.60E-2	1.59E-2	2.88E-1
Iron-59		8.97E-3	1.34E-2					2.24E-2
Cobalt-60		6.55E-2	1.15E-1	3.29E-1	1.52E-1	1.19E-1	1.15E-1	8.96E-1
Zinc-65		4.03E-3	7.78E-3	1.07E-2	7.68E-3	1.22E-2	6.77E-3	4.92E-2
Strontium-89		2.12E-3	1.33E-3	1.58E-3	1.52E-3	2.70E-4	7.55E-5	6.90E-3
Strontium-90		2.77E-4	3.08E-4	6.32E-4	6.46E-4	1.05E-4	6.44E-5	2.03E-3
Antimary-124		1.36E-3	1.31E-4	1.07E-3	7.17E-4	5.40E-4		3.82E-3
Iodine-131		9.07E-2	6.56E-3					9.73E-2
Cesium-134		8.06E-2	1.78E-1	4.71E-2	5.76E-2	2.04E-2	6.47E-2	4.48E-1
Cesium-137		2.27E-1	6.16E-1	1.45E-1	1.82E-1	5.56E-2	2.28E-1	1.45E0
Barium/Lanthanum-140		5.60E-5						5.60E-5
Neptunium-239		4.17E-4						4.17E-4
Unidentified		9.81E-2	4.44E-2	4.03E-3	1.40E-2	1.35E-4	2.28E-2	1.83E-1
8. Percent of Tech Spec Limit for Total Activity Released**		-	-	25.12	-	-	12.73	18.95

*No detectable activity
**Based on 10 Ci/quarter limit

TABLE A-2

SEMIANNUAL SUMMARY OF RADIOACTIVE GASEOUS EFFLUENTS
JANUARY-JUNE, 1974

		Jan	Feb	Mar	Apr	May	June	Total
1. Total Noble Gases	(Ci)							
(a) Main Stack		4.63E0	*-2	*-2	*-2	*-2	*-2	4.63E0
(b) Reactor Building Vent		8.64E1	1.24E2	1.22E2	1.22E2	8.44E1	6.37E1	6.03E2
2. Total Halogens *-1	(Ci)							
(a) Main Stack		3.65E-5	*-2	*-2	*-2	*-2	*-2	3.65E-5
(b) Reactor Building Vent		2.39E-3	4.29E-4	1.04E-5	*-3	*-3	*-3	2.83E-3
3. Total Particulate Gross Beta-Gamma Radioactivity *-1	(Ci)							
(a) Main Stack		1.47E-6	*-2	*-2	*-2	*-2	*-2	1.47E-6
(b) Reactor Building Vent		1.89E-3	4.49E-4	3.73E-4	2.66E-4	1.85E-4	6.99E-4	3.86E-3
4. Total Particulate Gross Alpha Radioactivity	(Ci)							
(a) Main Stack		2.42E-8	*-2	*-2	*-2	*-2	*-2	
(b) Reactor Building Vent		2.40E-7	*-4	*-4	6.12E-8	*-4	*-4	
5. Total Tritium	(Ci)							
(a) Main Stack		1.40E-2	*-2	*-2	*-2	*-2	*-2	1.40E-2
(b) Reactor Building Vent		4.96E-1	1.10E0	2.44E-1	1.08E-1	1.10E-1	2.25E-2	2.08E0
6. Maximum 24-hr Noble Gas Releases	(Date)	1-2	2-21	3-29	4-19	5-16	6-7	
	(Ci)	8.52	5.14	4.56	4.88	3.37	2.62	
7. Percent of Applicable Limit for Noble Gases		0.03	0.05	0.05	0.05	0.03	0.03	0.04
8. Percent of Applicable Limit for Halogens and Particulates		2.62	0.60	0.23	0.17	0.11	0.44	0.70
9. Isotopes Released	(Ci)							
A. Halogens								
Iodine-131		2.43E-3	4.29E-4	1.04E-5	*-3	*-3	*-3	2.87E-3
Iodine-133		*-3	*-4	*-4	*-3	*-4	*-4	
Iodine-135		*-3	*-4	*-4	*-3	*-4	*-4	
B. Particulates								
Beryllium-7				2.21E-5	3.27E-5			5.48E-5
Manganese-54		1.83E-4	8.09E-5	1.02E-4	6.08E-6	1.25E-4	3.81E-4	8.78E-4
Cobalt-58		1.15E-4	3.47E-5	3.65E-5	2.90E-6	4.11E-5	2.99E-5	2.60E-4
Iron-59		5.68E-5	9.35E-6	2.04E-5				8.66E-5
Cobalt-60		2.79E-4	9.63E-5	1.44E-4	9.08E-6	2.12E-4	2.63E-4	1.00E-3
Zinc-65		1.64E-5	1.06E-5	9.35E-6	1.18E-6	7.69E-7	9.17E-6	4.75E-5
Zirconium-95		2.51E-5	8.67E-6	6.29E-6	4.28E-6	1.11E-6		4.54E-5
Niobium-95						6.33E-5	1.42E-6	6.47E-6
Ruthenium-103		1.92E-5	7.90E-6		1.84E-6			2.89E-5
Ruthenium-106					5.37E-7			5.37E-7
Cesium-134			2.22E-6		5.98E-7	6.71E-5	1.43E-5	8.42E-5
Cesium-137		3.21E-6	5.63E-6	3.57E-6	2.79E-6	9.44E-5	6.10E-5	1.75E-4
Barium-Lanthanum-140		9.93E-4						9.93E-4
Cerium-141		2.50E-5	4.43E-6	2.55E-6				3.20E-5
Cerium-144		3.08E-5	1.16E-5	1.19E-5	6.85E-6	2.73E-6	3.72E-6	6.76E-5
Strontium-89		<1.76E-6	*-4	*-4	<6.27E-7	*-4	*-4	
Strontium-90		<2.69E-7	*-4	*-4	6.17E-7	*-4	*-4	
C. Gases *-5		*-2	*-2	*-2	*-2	*-2	*-2	

*-1 With half-lives greater than 8 days.
*-2 Unit out for refueling and maintenance: main stack out of service.
*-3 No detectable activity.
*-4 Quarterly analysis - Tech Specs.
*-5 Main stack only.

BOSTON EDISON COMPANY

PILGRIM NUCLEAR GENERATING STATION

Environmental Radiation Monitoring Program

SEMIANNUAL REPORT NO. 5
JULY 1, 1974 THROUGH DECEMBER 31, 1974

Prepared By

Joel I. Cehn
Environmental Sciences Group
Nuclear Engineering Department

March 1974

Approved By: G. James Davis

G. James Davis, Manager
Environmental Sciences Group

I. INTRODUCTION AND SUMMARY

This report describes the data accumulated in the Environmental Radiation Surveillance Program during the semiannual period July 1 through December 31, 1974.

After a seven-month outage for refueling, maintenance, and contested licensing proceedings, the unit came up to full power on August 16. Plant capacity factors (a measure of electrical output) during the reporting period were: July, 5 percent, August, 84 percent; September, 73 percent; October, 83 percent; November, 82 percent; and December, 75 percent.*

On December 17, an augmented off-gas treatment system was put into service. This system will effectively eliminate iodines, and greatly reduce the amount of radiogases released from the main stack. Also put into service was a liquid waste solidification system, which is expected to have the effect of significantly reducing the quantity of liquid wastes discharged to the bay. A summary of radioactive effluents released during the reporting period is presented in Appendix A.

The liquid releases associated with and occuring during the seven-month outage (prior to completion of the solidification system) are evident in the program data reported in Section III of this report. Media in the vicinity of the station found with trace amounts of radioactivity from the station were Irish moss, bottom sediment, mussels, lobster (1 individual), cod (2), herring (2), bluefish (2) and cunner. A potential radiation dose to humans consuming these media was calculated using "worst case" assumptions. This dose was calculated to be 0.26 millirem per year resulting from station produced radioactivity and 6.7

*Details of plant operation can be found in a separate report entitled, "Operating and Maintenance Semiannual Report #5."

millirem from nuclear weapons test fallout radioactivity. These results may be compared to the natural background radiation levels of approximately 100 millirem per year. No radioactivity attributable to station operation was detected in any terrestrial media.

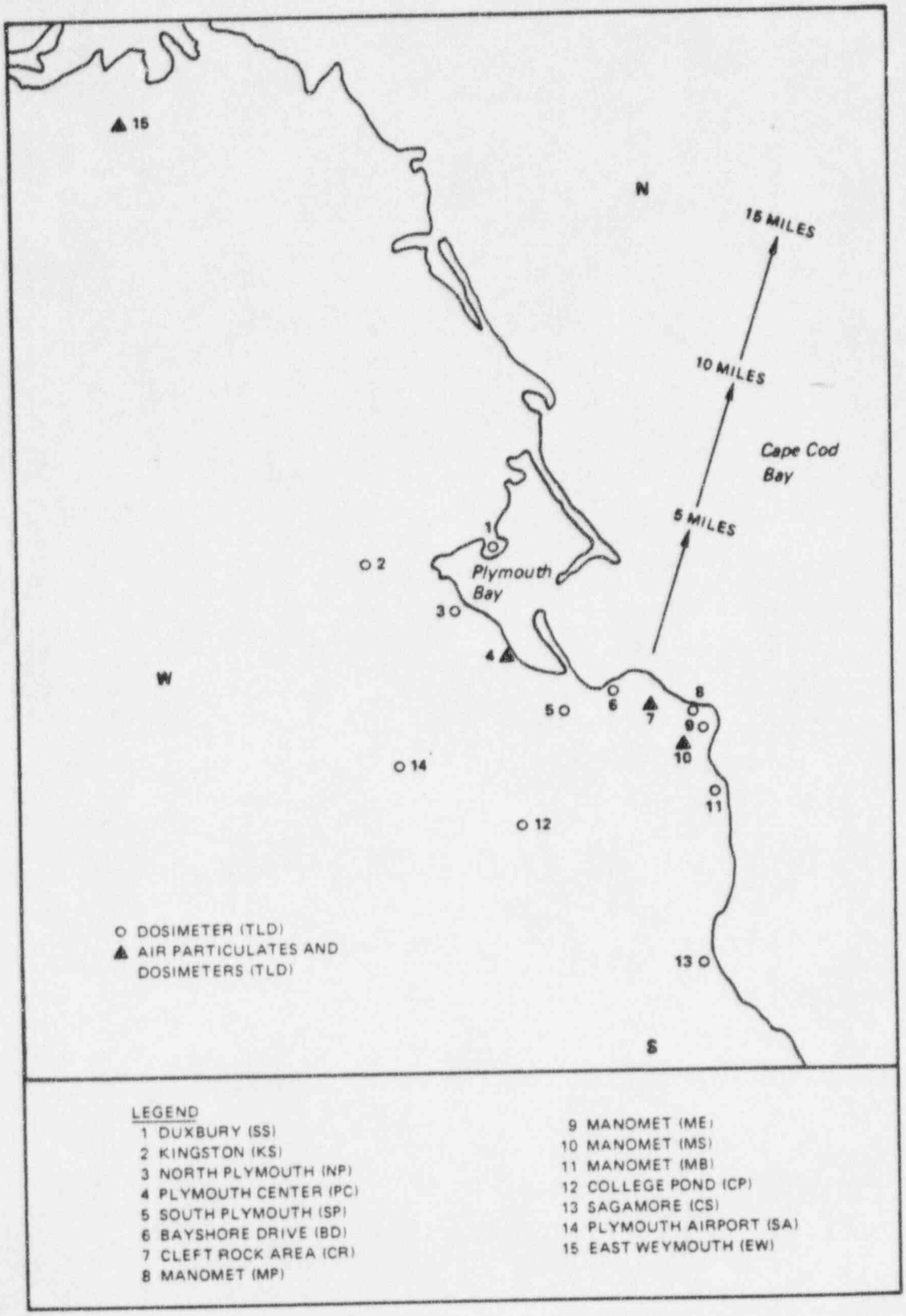

Figure 1. Location of Offsite Radiological Monitoring Stations

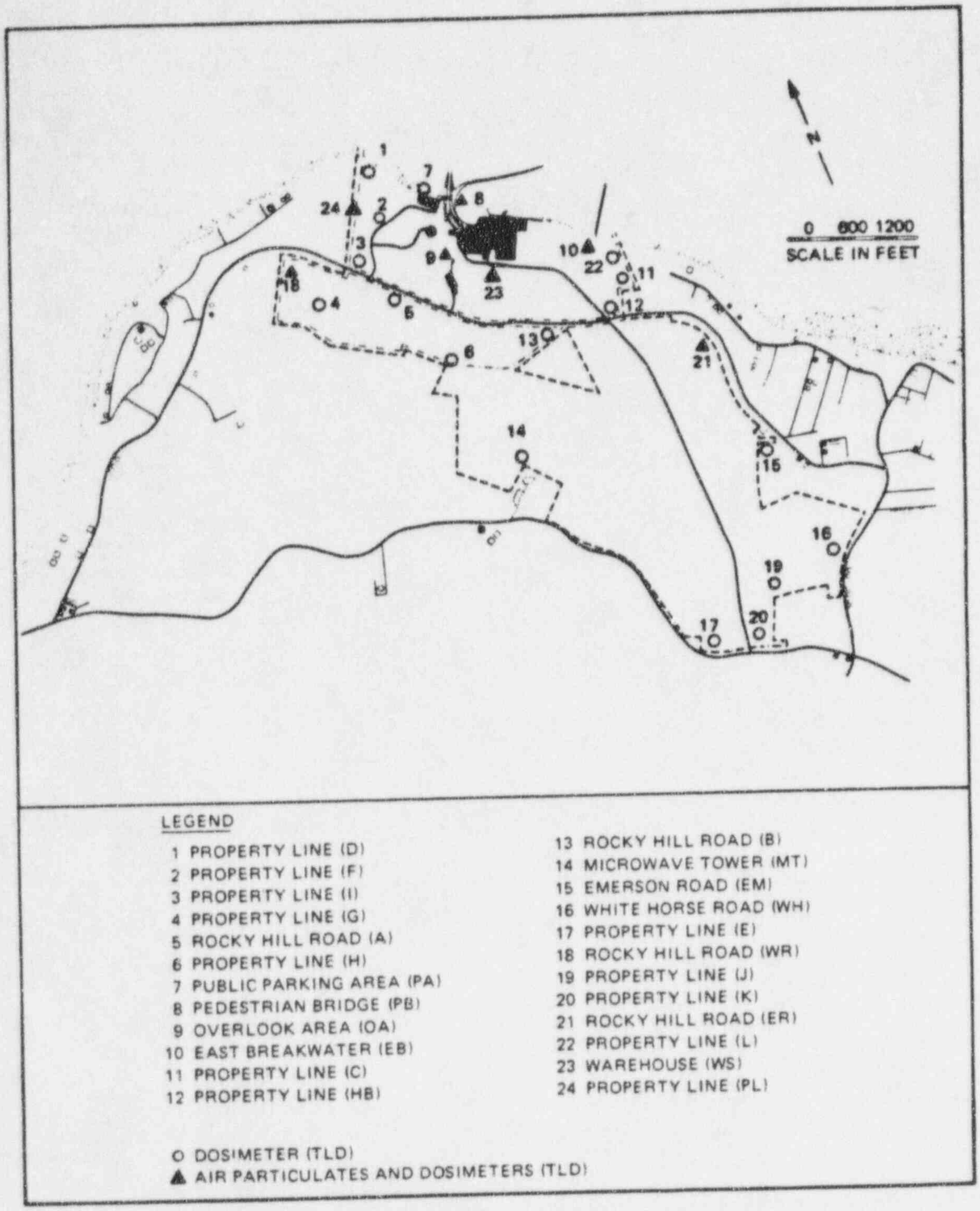

Figure 2. Location of Onsite Radiological Monitoring Stations

TABLE 7A

AIR PARTICULATES - GAMMA ISOTOPE CONCENTRATION IN MONTHLY COMPOSITES (pCi/m^3)[a]

	Control	Offsite			Onsite						
Collection Period (1974)	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property Line	Overlook Area	Pedestrian Bridge	East Breakwater	Rocky Hill Road (East)	Warehouse
September											
K-40	0.10 ±0.08	0.2 ±0.2	0.09 ±0.07	(b)	(b)	(b)	(b)	(b)	(b)	(b)	0.11 ±0.07
Th-228	(b)	0.2 ±0.2	0.16 ±0.009	(b)	(b)	0.012 ±0.009	(b)	(b)	(b)	(b)	(b)
Ru-106	(b)	(b)	(b)	(b)	(b)	(b)	(b)	(b)	(b)	(b)	0.03 ±0.03
October											
K-40	(b)	(b)	(b)	0.08 ±0.04	(b)	(b)	0.05 ±0.05	(b)	(b)	(b)	(b)
Th-228	(b)	(b)	(b)	(b)	(b)	(b)	(b)	(b)	(b)	(b)	0.009 ±0.008
Ru-106	(b)	(b)	(b)	(b)	(b)	(b)	(b)	(b)	0.03 ±0.03	0.04 ±0.02	(b)

(a) Results of Ge(Li) spectrometry. No isotopes other than those listed in table were detected. Analysis required quarterly.

(b) Less than MDA (Minimum Detectable Activity). Typical MDA's are:

K-40	0.05 pCi/m^3
Th-228	0.01
Ru-106	0.03

TABLE 7B

AIR PARTICULATES - STRONTIUM-90 CONCENTRATION IN QUARTERLY COMPOSITES ($pCi/m^3 \times 10^{-4}$)[a][b]

	Control	Offsite			Onsite						
Collection Period (1974)	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property Line	Overlook Area	Pedestrian Bridge	East Breakwater	Rocky Hill Road (East)	Warehouse
Jan - Mar	2 ±1	6 ±1	4 ±1	2 ±1	1 ±1	4 ±1	<1	2 ±1	5 ±4	<2	3 ±1
Apr - Jun	4 ±1	7 ±2	5 ±1	2 ±1	4 ±2	6 ±1	6 ±1	6 ±1	3 ±1	5 ±1	3 ±1
July - Sept	(c)	7 ±1	<1	5 ±1	8 ±1	3 ±1	6 ±1	5 ±1	3 ±1	3 ±1	3 ±1
Oct - Dec	4 ±1	2 ±1	(c)	3 ±1	5 ±1	4 ±1	1 ±1	2 ±1	3 ±1	2 ±1	2 ±1

(a) 3.0 in table means 3×10^{-4} pCi/m^3

(b) These results are reported with an error corresponding to two standard deviations in the counting error. At the low count rates encountered, it is difficult to verify the half-life of Y-90 and the actual error could be larger than that reported.

(c) Sample lost in analysis.

TABLE 8

PARTICULATE IODINE-131 IN AIR SAMPLES (pCi/m^3)

	Control	Offsite			Onsite						
Collection Period	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property Line*	Overlook Area	Pedestrian Bridge*	East Breakwater*	Rocky Hill Road (East)	Warehouse*
June 13 - June 20	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
June 20 - June 27	<1	<1	<1	<1	<1	<1	<1	<1	(a)	<1	<1
June 27 - July 4	<1	<1	<1	<1	<1	<1	(a)	<1	(a)	<1	<1
July 4 - July 11	<1	<1	<1	<1	<1	<1	<1	<1	(a)	<1	<1
July 11 - July 13	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
July 18 - July 25	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
July 25 - Aug 1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Aug 1 - Aug 8	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Aug 8 - Aug 15	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Aug 15 - Aug 22	<1	(a)	<1	<1	<1	<1	<1	<1	<1	<1	<1
Aug 22 - Aug 29	<1	(a)	<1	<1	<1	<1	<1	<1	<1	<1	<1
Aug 29 - Sept 5	<1	(a)	<1	<1	<1	<1	<1	<1	<1	<1	<1
Sept 5 - Sept 12	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Sept 12 - Sept 19	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Sept 19 - Sept 26	<1	(a)	<1	<1	<1	<1	<1	<1	<1	<1	<1
Sept 26 - Oct 3	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Oct 3 - Oct 10	<1	<1	<1	<1	<1	<1	<1	(a)	<1	<1	<1
Oct 10 - Oct 17	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Oct 17 - Oct 24	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Oct 24 - Oct 31	<1	<1	<1	<1	<1	<1	<1	<1	(a)	<1	<1
Oct 31 - Nov 7	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Nov 7 - Nov 14	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Nov 14 - Nov 21	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Nov 21 - Nov 28	<1	<1	<1	<1	<1	(a)	<1	<1	<1	<1	<1
Nov 28 - Dec 5	<1	<1	<1	<1	<1	(a)	<1	<1	<1	<1	<1
Dec 5 - Dec 12	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Dec 12 - Dec 19	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Dec 19 - Dec 26	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Dec 26 - Jan 2	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1

(a) No sample.
* Station not required by operating license.

TABLE 8A

GASEOUS IODINE-131 IN AIR SAMPLES (pCi/m^3)

	Control	Offsite			Onsite						
Collection Period	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property Line*	Overlook Area	Pedestrian Bridge*	East Breakwater*	Rocky Hill Road (East)	Warehouse*
June 20 - June 27	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
June 27 - July 4	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
July 4 - July 11	<1	<1	<1	<1	<1	<1	<1	<1	(a)	<1	<1
July 11 - July 18	<1	<1	<1	<1	<1	<1	<1	<1	(a)	<1	<1
July 18 - July 25	<1	<1	<1	<1	<1	<1	<1	<1	(a)	<1	<1
July 25 - Aug 1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Aug 1 - Aug 8	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Aug 8 - Aug 15	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Aug 15 - Aug 22	<1	(a)	<1	<1	<1	<1	<1	<1	<1	<1	<1
Aug 22 - Aug 29	<1	(a)	<1	<1	<1	<1	<1	<1	<1	<1	<1
Aug 29 - Sept 5	<1	(a)	<1	<1	<1	<1	<1	<1	<1	<1	<1
Sept 5 - Sept 12	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Sept 12 - Sept 19	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Sept 19 - Sept 26	<1	(a)	<1	<1	<1	<1	<1	<1	<1	<1	<1
Sept 26 - Oct 3	<1	<1	<1	<1	<1	<1	<1	(a)	<1	<1	<1
Oct 3 - Oct 10	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Oct 10 - Oct 17	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Oct 17 - Oct 24	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Oct 24 - Oct 31	<1	<1	<1	<1	<1	<1	<1	<1	(a)	<1	<1
Oct 31 - Nov 7	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Nov 7 - Nov 14	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Nov 14 - Nov 21	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Nov 21 - Nov 28	<1	<1	<1	<1	<1	(a)	<1	<1	<1	<1	<1
Nov 28 - Dec 5	<1	<1	<1	<1	<1	(a)	<1	<1	<1	<1	<1
Dec 5 - Dec 12	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Dec 12 - Dec 19	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Dec 19 - Dec 26	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1
Dec 26 - Jan 2	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1	<1

(a) No sample.
(b) Sample lost during laboratory move.
* Station not required by operating license.

TABLE A-1

SEMI-ANNUAL SUMMARY OF RADIOACTIVE LIQUID EFFLUENTS
JULY - DECEMBER 1974

	July	Aug.	Sept.	Oct.	Nov.	Dec.	Total
1. Gross Radioactivity							
(a) Total Released (Ci)	1.28 E-1	4.87 E-3	1.23 E-2	2.18 E-2	1.64 E-1	1.03 E-1	4.34 E-1
(b) Avg. Conc. Released (uCi/ml)	4.32 E-7	1.78 E-8	3.73 E-8	2.65 E-8	5.45 E-8	4.09 E-8	5.99 E-8
(c) Max. Conc. Released (uCi/ml)	8.24 E-7	3.17 E-8	1.56 E-7	1.35 E-7	1.91 E-7	9.78 E-8	8.24 E-7
2. Tritium							
(a) Total Released (Ci)	1.15 E-1	2.42 E-4	6.71 E-5	6.75 E-3	1.99 E-2	3.33 E-2	1.75 E-1
(b) Avg. Conc. Released (uCi/ml)	3.89 E-7	8.86 E-10	2.03 E-10	8.20 E-9	6.61 E-9	1.32 E-9	2.41 E-8
3. Dissolved Noble Gases							
(a) Total Released (Ci)	(a)	(a)	(a)	(a)	(a)	(a)	(a)
(b) Avg. Conc. Released (uCi/ml)							
4. Gross Alpha Radioactivity							
(a) Total Released (Ci)	<7.05 E-5	<1.77 E-7	1.10 E-4	3.83 E-7	<1.81 E-7	<2.13 E-7	<1.81 E-4
(b) Avg. Conc. Released (uCi/ml)	<2.38 E-10	<6.48 E-13	3.33 E-10	4.65 E-13	<6.01 E-14	<8.45 E-14	<2.50 E-11
5. Volume of Liquid Waste to Discharge Canal (Liters)	2.35 E+5	1.97 E+4	9.16 E+3	1.74 E+4	1.81 E+4	1.65 E+4	3.16 E+5
6. Volume of Dilution Water (Liters)	2.96 E+8	2.73 E+8	3.30 E+8	8.23 E+8	3.01 E+9	2.52 E+9	7.25 E+9
7. Isotopes Released (Ci)							
Chromium-51			7.59 E-4	5.45 E-3	6.43 E-3	1.05 E-2	2.32 E-2
Manganese-54	1.23 E-2	1.04 E-3	1.71 E-3	1.10 E-3	4.31 E-3	2.58 E-3	2.30 E-2
Cobalt-58	4.11 E-3	1.18 E-4	1.64 E-3	3.36 E-4	1.14 E-3	5.90 E-4	7.93 E-3
Iron-59			3.67 E-5	3.05 E-4	2.40 E-4	8.42 E-5	6.66 E-4
Cobalt-60	4.54 E-2	1.98 E-3	3.84 E-3	2.04 E-3	2.77 E-3	3.40 E-3	5.94 E-2
Zinc-65	5.85 E-3	9.97 E-5	2.08 E-4	1.25 E-4	1.25 E-4	1.12 E-4	6.52 E-3
Zirconium/Niobium-95	8.60 E-4	1.35 E-5	3.58 E-5	1.10 E-4	6.56 E-6	3.37 E-6	1.03 E-3
Molybdenum-99/Technecium-99m					8.94 E-3	1.04 E-3	9.98 E-3
Silver-110m	1.01 E-4		2.18 E-5	5.42 E-5	1.83 E-5		1.95 E-4
Iodine-131	1.35 E-4	1.87 E-5	1.24 E-3	3.91 E-3	5.96 E-2	8.29 E-3	7.32 E-2
Iodine-133	4.53 E-4		1.98 E-6	6.36 E-5	1.43 E-4	4.79 E-4	1.14 E-3
Cesium-134	1.38 E-2	2.90 E-4	3.62 E-4	1.59 E-3	1.82 E-2	1.68 E-2	5.10 E-2
Cesium-136			9.10 E-5	2.65 E-4		9.07 E-4	1.26 E-3
Cesium-137	4.18 E-2	1.05 E-3	9.65 E-4	4.21 E-3	4.41 E-2	4.06 E-2	1.33 E-1
Barium/Lanthunum-140			9.38 E-4	3.63 E-4	3.23 E-3	8.79 E-4	5.41 E-3
Cerium-141			4.20 E-6	4.24 E-5	1.42 E-5	1.32 E-5	7.40 E-5
Cerium-144	7.84 E-4		1.98 E-5				8.04 E-4
Neptunium-239	9.01 E-4		1.96 E-4	4.79 E-5	7.58 E-4	4.27 E-4	2.33 E-3
Strontium-89	1.10 E-4	1.44 E-4	6.23 E-5	5.92 E-4	5.07 E-3	3.29 E-3	9.27 E-3
Strontium-90	1.27 E-4	9.46 E-6	9.16 E-6	2.96 E-5	3.08 E-4	1.99 E-4	6.82 E-4
Unidentified	1.07 E-3	1.10 E-4	1.30 E-4	1.12 E-3	8.78 E-3	1.29 E-2	2.41 E-2
8. Percent of Tech. Spec. Limit For Total Activity Released (b)			1.45			2.89	2.17

(a) No Detectable Activity
(b) Based on 10 Ci/quarter limit

TABLE A-2

SEMI-ANNUAL SUMMARY OF RADIOACTIVE GASEOUS EFFLUENTS
JULY - DECEMBER 1974

		July	Aug.	Sept.	Oct.	Nov.	Dec.	Total
1. Total Noble Gases	(Ci)							
(a) Main Stack		1.31 E+3	2.31 E+4	7.53 E+4	1.20 E+5	2.50 E+5	6.77 E+4	5.37 E+5
(b) Reactor Building Vent		4.81 E+1	5.11 E+2	5.72 E+2	1.04 E+3	2.50 E+3	3.89 E+3	8.55 E+3
2. Total Halogens *-1	(Ci)							
(a) Main Stack		6.41 E-4	4.27 E-2	9.58 E-2	2.14 E-1	8.38 E-1	9.65 E-2	1.29 E 0
(b) Reactor Building Vent		1.51 E-6	6.26 E-4	7.68 E-3	7.81 E-3	5.38 E-2	8.33 E-2	1.53 E-1
3. Total Particulate Gross Beta-Gamma Radioactivity *-1	(Ci)							
(a) Main Stack		2.40 E-4	1.88 E-3	1.02 E-3	1.50 E-3	1.11 E-3	3.83 E-4	6.13 E-3
(b) Reactor Building Vent		2.59 E-4	7.76 E-4	9.58 E-4	5.00 E-4	2.70 E-3	3.55 E-3	8.74 E-3
4. Total Particulate Gross Alpha Radioactivity	(Ci)							
(a) Main Stack		*-2	*-2	5.38 E-9	*-2	6.99 E-8	*-2	
(b) Reactor Building Vent		*-2	*-2	3.18 E-9	*-2	1.54 E-7	*-2	
5. Total Tritium	(Ci)							
(a) Main Stack		1.70 E-2	1.[illegible]4 E-1	1.82 E-1	2.18 E-1	3.29 E-1	3.73 E-1	1.25 E 0
(b) Reactor Building Vent		2.50 E-2	3.83 E-1	2.04 E-1	6.85 E-1	3.08 E-1	3.03 E 0	4.63 E 0
6. Maximum 24-hr Noble	(Date)	7-31	8-27	9-29	10-31	11-5	12-1	
Gas Release	(Ci)	4.03 E+2	1.46 E+3	6.20 E+3	7.53 E+3	1.24 E+4	8.26 E+3	
7. Percent of Applicable Limit For Noble Gases		0.21	3.64	11.82	18.28	39.56	11.57	14.02
8. Percent of Applicable Limit For Halogens & Particulates		0.19	2.20	8.48	11.57	61.84	56.09	23.25
9. Isotopes Released	(Ci)							
A. Halogens								
Iodine-131		6.43 E-4	4.33 E-2	1.03 E-1	2.22 E-1	8.92 E-1	1.80 E-1	1.44 E 0
Iodine-133		*-2	*-2	3.15 E-1	*-2	*-2	8.09 E-1	
Iodine-135		*-2	*-2	8.06 E-1	*-2	*-2	1.06 E 0	
B. Particulates								
Chromium-51						1.35 E-5		1.35 E-5
Manganese-54		4.52 E-5	2.69 E-5	3.15 E-5	9.85 E-6	8.01 E-5	1.02 E-5	1.13 E-4
Cobalt-58		8.54 E-6		3.34 E-6		2.99 E-5	5.75 E-6	4.76 E-5
Iron-59						1.13 E-5		1.13 E-5
Cobalt-60		1.01 E-4	5.23 E-5	7.79 E-5	2.74 E-5	7.75 E-5	4.04 E-5	3.77 E-4
Zinc-65		4.20 E-6				4.74 E-7		4.67 E-6
Zr-Nb-95						3.33 E-6		3.33 E-6
Silver-110m							2.68 E-6	2.68 E-6
Antimony-124							1.95 E-5	1.95 E-5
Cesium-134		4.54 E-6		2.39 E-6		2.49 E-5	3.58 E-5	6.76 E-5
Cesium-137		2.14 E-5	1.95 E-5	2.75 E-5	1.26 E-5	7.56 E-5	1.12 E-4	2.68 E-4
Cerium-139							1.94 E-3	1.94 E-3
Barium-Lanthunum-140			5.66 E-4	3.35 E-4	5.72 E-4	5.43 E-4	1.69 E-3	3.71 E-3
Cerium-141					2.96 E-7	7.43 E-6	1.52 E-5	2.29 E-5
Cerium-144			3.94 E-6			5.37 E-6		9.31 E-6
Strontium-89		*-2	*-2	2.52 E-4	*-2	6.32 E-4	*-2	
Strontium-90		*-2	*-2	1.37 E-6	*-2	3.26 E-6	*-2	
C. Gases *-3								
Xenon-138		2.90 E+1	3.18 E+3	1.10 E+4	1.61 E+4	2.74 E+4	5.27 E+3	6.30 E+4
Krypton-87		2.29 E+2	2.97 E+3	8.45 E+3	1.61 E+4	3.49 E+4	8.52 E+3	7.12 E+4
Krypton-88		2.37 E+2	2.50 E+3	1.14 E+4	2.07 E+4	4.50 E+4	1.10 E+4	9.08 E+4
Krypton-85m		8.08 E+1	9.84 E+2	3.91 E+3	7.45 E+3	1.91 E+4	4.73 E+3	3.63 E+4
Xenon-135		3.28 E+2	3.87 E+3	1.71 E+4	3.18 E+4	7.82 E+4	2.17 E+4	1.53 E+5
Xenon-133		6.59 E+1	2.70 E+3	1.28 E+4	1.84 E+4	3.58 E+4	1.16 E+4	8.14 E+4
Sum of Remainder		3.40 E+2	6.90 E+3	1.06 E+4	9.45 E+3	9.60 E+3	4.88 E+3	4.18 E+4

*-1 With half-lives greater than 8 days
*-2 Quarterly analysis-Tech. Specs.
*-3 Main stack only.

BOSTON EDISON COMPANY

PILGRIM NUCLEAR GENERATING STATION

Environmental Radiation Monitoring Program

SEMIANNUAL REPORT NO. 6
JANUARY 1, 1975 THROUGH JUNE 30, 1975

Prepared By

Joel I. Cehn
Environmental Sciences Group
Nuclear Engineering Department

September 1975

Approved By: G. James Davis

G. James Davis, Manager
Environmental Sciences Group

I. INTRODUCTION AND SUMMARY

This report describes the data accumulated in the Environmental Radiation Surveillance Program during the semiannual period January 1 through June 30, 1975.

Plant operation during the reporting period is detailed in a separate report entitled "Operating and Maintenance Semiannual Report No. 6". Power levels during this period were limited, administratively, to 80 percent of full power from February through May and 70 percent of full power during June. These limits on power level were set by Boston Edison Company for reasons of operation and maintenance. Plant capacity factors (a measure of electrical output) during the reporting period were: January, 36 percent; February, 34 percent; March 76 percent; April, 42 percent; May, 55 percent; and June, 65 percent.

The Environmental Radiation Surveillance Program was implemented during the reporting period as required by the Nuclear Regulatory Commission. Plant-related radioactivity levels decreased from 1974 levels in molluscs, algae, and sediment. The current levels of Mn-54, Co-60 and Cs-137 in these media are less than 1 picocurie per gram.

Weapons test fallout was detected in air particulates, specifically Nb-95, Ce-144 and Cs-137. These elements were also detected in various other media. Gaseous radio-iodine was detected onsite at levels less than 1 picocurie per cubic meter. This I-131 was attributed to releases from the reactor building vent.

Finally, marine media data from 1974 were reviewed and calculations showed that the seafood ingestion dose to an individual was less than 1 millirem per year.

TABLE 6

AIR PARTICULATES - GROSS BETA CONCENTRATIONS IN WEEKLY SAMPLES (pCi/m^3)

Sampling Period (1975)	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property Line*	Overlook Area	Pedestrian Bridge*	East Breakwater	Rocky Hill Road (East)	Warehouse
Jan 2 - Jan 9	0.05	0.05	0.04	0.04	0.08	0.08	0.08	0.07	0.09	0.04	0.08
Jan 9 - Jan 16	0.10	0.10	0.09	0.11	0.10	0.11	0.09	0.11	0.10	0.09	0.11
Jan 16 - Jan 23	0.11	0.13	0.09	0.10	0.12	0.09	0.11	0.10	0.11	0.10	0.09
Jan 23 - Jan 30	0.10	0.09	0.09	0.08	0.13	0.10	0.10	0.10	0.09	0.09	0.11
Jan 30 - Feb 6	(a)	0.09	0.08	0.07	0.10	(a)	(a)	(b)	(a)	0.08	0.09
Feb 6 - Feb 13	(a)	0.13	0.09	0.07	0.09	(a)	(a)	(a)	(a)	0.09	0.09
Feb 13 - Feb 18	0.07	0.12	0.07	0.06	0.08	0.06	0.07	0.07	0.06	0.06	0.10
Feb 18 - Feb 27	0.10	0.11	0.11	0.04	0.11	0.10	0.10	0.09	0.11	0.11	0.12
Feb 27 - Mar 6	0.11	0.12	0.09	0.02	0.11	0.12	0.09	0.10	0.14	0.10	0.11
Mar 6 - Mar 13	0.12	0.13	0.10	(b)	0.14	0.03	0.11	0.10	0.13	0.12	0.17
Mar 13 - Mar 20	0.14	0.09	0.13	0.11	0.15	0.13	0.17	0.13	0.13	0.10	0.19
Mar 20 - Mar 27	0.10	0.14	0.11	0.12	0.13	0.13	0.14	0.11	0.13	(b)	(b)
Mar 27 - Apr 3	0.03	0.18	0.05	0.13	0.18	0.16	0.13	0.15	0.15	0.16	0.15
Apr 3 - Apr 10	0.06	0.06	0.07	0.06	0.07	0.07	0.05	0.06	0.08	0.08	0.16
Apr 10 - Apr 17	0.20	0.20	0.19	0.18	0.17	0.21	0.18	0.21	0.22	0.19	0.20
Apr 17 - Apr 24	0.19	0.17	0.13	0.19	0.18	0.14	0.20	0.18	0.14	0.13	0.19
Apr 24 - May 1	0.17	0.07	0.06	0.08	0.07	0.05	0.05	0.07	0.06	0.04	0.08
May 1 - May 8	0.12	0.11	0.10	0.11	0.12	0.12	0.12	0.13	0.12	0.07	0.15
May 8 - May 15	0.08	0.09	0.08	0.09	0.07	0.08	(b)	0.08	0.08	(b)	0.06
May 15 - May 22	0.13	0.11	0.13	0.10	0.13	0.12	0.15	0.15	0.17	0.14	0.13
May 22 - May 29	0.07	0.07	0.07	0.05	0.06	0.06	0.04	0.05	0.06	0.06	0.03
May 29 - June 5	0.07	0.08	0.08	0.07	0.08	0.08	0.05	0.07	0.04	0.07	0.07
June 5 - June 12	0.06	0.06	0.07	0.02	0.07	0.09	0.06	0.07	0.07	0.06	0.06
June 12 - June 19	0.06	0.08	0.05	0.05	0.06	0.07	0.02	0.06	0.06	0.07	0.07
June 19 - June 26	0.07	0.07	0.05	0.08	0.09	0.10	0.03	0.09	0.08	0.08	0.09

(a) Station inaccessable - no sample until following week.
(b) Instrument out of service.

TABLE 7

AIR PARTICULATES - GROSS GAMMA CONCENTRATION IN MONTHLY COMPOSITES
$(cpm/m^3 \times 10^{-3})$ (a)

	Control	Offsite			Onsite						
Collection Period (1975)	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property Line	Overlook Area	Pedestrian Bridge	East Breakwater	Rocky Hill Road (East)	Warehouse
Jan	5	6	< 3	3	9	3	4	10	6	4	7
Feb	19	19	17	14	23	7	14	8	16	12	32
Mar	21	15	17	14	28	20	25	19	19	8	12
Apr	15	26	15	22	26	22	30	25	28	36	22
May	12	15	16	14	12	11	11	15	16	9	8
June	20	13	14	15	18	16	6	14	21	9	20

(a) Error is ± 2 or 10%, whichever is larger.

TABLE 7A

AIR PARTICULATES - GAMMA ISOTOPE CONCENTRATION IN MONTHLY COMPOSITES
(pCi/m^3) (a)

Collection Period (1975)	Control	Offsite			Onsite						
	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property Line	Overlook Area	Pedestrian Bridge	East Breakwater	Rocky Hill Road (East)	Warehouse
January											
Nb-95	0.013 ±0.007	0.018 ±0.007	(b)	0.022 ±0.007	0.026 ±0.007	0.024 ±0.007	0.013 ±0.007	(b)	0.018 ±0.007	0.015 ±0.007	0.015 ±0.007
April											
Nb-95	0.013 ±0.006	0.011 ±0.008	(b)	(b)	0.009 ±0.005	0.013 ±0.006	(b)	(b)	(b)	(b)	(b)
Cs-137	(b)	0.003 ±0.002	(b)	0.004 ±0.002	(b)	(b)	(b)	0.004 ±0.002	(b)	(b)	(b)
Ce-144	(b)	0.051 ±0.021	(b)	0.044 ±0.017	(b)	(b)	(b)	0.035 ±0.012	(b)	(b)	(b)

(a) Results of Ge(Li) spectrometry. Analysis required quarterly. Nominal MDA's (minimum detectable activities) for these isotopes are:

Nb-95 - 0.006 pCi/m^3
Ce-144- 0.03
Cs-137- 0.007

(b) Less than MDA.

TABLE 8

PARTICULATE IODINE-131 IN AIR SAMPLES (pCi/m^3)
JANUARY - JUNE 1975

	Control	Offsite			Onsite						
Collection Period	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property Line	Overlook Area	Pedestrian Bridge	East Breakwater	Rocky Hill Road (East)	Warehouse
Jan 2 - Jan 9	0.03±0.03	<0.03	<0.03	<0.02	<0.02	<0.02	<0.03	<0.02	<0.02	<0.02	<0.03
Jan 9 - Jan 16	<0.01	<0.01	<0.01	<0.01	0.01±0.01	0.01±0.01	<0.01	<0.01	<0.02	<0.01	<0.01
Jan 16 - Jan 23	<0.09	0.09±0.09	<0.09	<0.08	0.11±0.09	<0.08	<0.09	<0.08	0.09±0.09	<0.08	0.12±0.10
Jan 23 - Jan 30	<0.03	<0.03	<0.03	<0.02	0.02±0.02	<0.02	<0.03	<0.02	0.03±0.03	<0.03	<0.03
Jan 30 - Feb 6	(a)	<0.03	<0.03	<0.02	0.02±0.02	(a)	(a)	(b)	(a)	<0.03	<0.03
Feb 6 - Feb 13	(a)	<0.03	<0.04	0.04±0.03	0.04±0.03	(a)	(a)	(a)	(a)	<0.02	<0.02
Feb 13 - Feb 20	0.02±0.02	0.06±0.04	<0.03	<0.03	<0.03	<0.01	0.02±0.02	0.02±0.02	<0.02	<0.03	<0.04
Feb 20 - Feb 27	<0.02	0.02±0.02	<0.02	<0.02	<0.02	<0.02	<0.02	0.01±0.01	<0.02	<0.02	0.02±0.02
Feb 27 - Mar 6	0.04±0.04	0.03±0.03	0.04±0.04	<0.03	0.03±0.03	<0.03	<0.05	0.03±0.03	0.04±0.03	<0.03	<0.04
Mar 6 - Mar 13	<0.03	0.03±0.03	<0.03	(b)	0.05±0.03	<0.02	<0.03	<0.02	<0.03	<0.03	0.04±0.03
Mar 13 - Mar 20	<0.02	0.02±0.02	0.04±0.03	<0.03	0.03±0.02	<0.02	0.02±0.02	<0.02	0.03±0.02	<0.02	<0.12(c)
Mar 20 - Mar 27	<0.02	<0.02	0.02±0.02	<0.02	<0.02	<0.02	0.02±0.02	0.02±0.02	0.04±0.02	(b)	(b)
Mar 27 - Apr 3	<0.02	0.02±0.02	0.02±0.02	0.04±0.02	0.03±0.02	0.02±0.02	0.02±0.02	0.02±0.02	<0.02	0.02±0.02	0.03±0.02
Apr 3 - Apr 10	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	0.02±0.02	0.05±0.02
Apr 10 - Apr 17	0.06±0.03	0.05±0.04	0.05±0.04	0.03±0.03	0.05±0.03	0.05±0.03	0.05±0.03	0.03±0.03	0.05±0.03	0.07±0.03	0.07±0.04
Apr 17 - Apr 24	<0.02	<0.02	<0.02	0.03±0.02	0.04±0.02	<0.02	0.04±0.02	0.02±0.02	0.04±0.02	0.03±0.02	0.03±0.02
Apr 24 - May 1	<0.01	<0.02	0.08±0.04 (c)	0.01±0.01	0.03±0.01	0.02±0.01	0.01±0.01	0.01±0.01	0.01±0.01	0.02±0.01	0.02±0.01
May 1 - May 8	0.02±0.01	0.01±0.01	0.02±0.01	<0.01	0.02±0.01	0.01±0.01	0.02±0.01	0.01±0.01	0.02±0.01	<0.01	0.02±0.01
May 8 - May 15	<0.02	<0.02	0.02±0.02	0.02±0.02	0.02±0.02	0.02±0.02	(b)	0.02±0.02	<0.02	(b)	<0.02
May 15 - May 22	0.03±0.02	<0.02	<0.02	0.01±0.01	0.01±0.01	<0.01	0.02±0.02	0.02±0.02	<0.01	<0.02	<0.01
May 22 - May 29	0.03±0.03	<0.03	<0.03	0.02±0.02	<0.02	<0.02	<0.03	0.04±0.03	<0.02	0.02±0.02	<0.03
May 29 - June 5	0.02±0.01	0.03±0.02	0.02±0.02	0.02±0.01	0.02±0.01	0.03±0.01	0.03±0.02	0.02±0.02	0.03±0.02	0.03±0.01	0.02±0.01
June 5 - June 12	<0.02	0.03±0.02	0.10±0.04 (c)	<0.01	0.02±0.02	<0.02	<0.02	<0.02	<0.02	0.02±0.02	0.02±0.02
June 12 - June 19	<0.02	<0.02	<0.02	<0.02	0.02±0.02	<0.02	<0.02	<0.02	<0.02	0.02±0.02	<0.02
June 19 - June 26	<0.02	<0.01	<0.01	<0.01	<0.01	0.01±0.01	<0.01	<0.01	<0.01	<0.01	0.01±0.01

(a) Station inaccessable - no sample until following week.
(b) Instrument out of service.
(c) Instrument malfunction - partial sample.

TABLE 8A

GASEOUS IODINE-131 IN AIR SAMPLES (pCi/m^3)
JANUARY - JUNE 1975

	Control	Offsite			Onsite						
Collection Period	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property Line	Overlook Area	Pedestrian Bridge	East Breakwater	Rocky Hill Road (East)	Warehouse
Jan 1 - Jan 9	<0.02	<0.02	0.02±0.02	<0.02	<0.02	<0.02	<0.02	<0.02	0.05±0.02	0.03±0.02	0.05±0.03
Jan 9 - Jan 16	<0.02	<0.02	<0.03	<0.02	<0.02	<0.02	<0.02	<0.02	<0.03	<0.02	0.04±0.03
Jan 16 - Jan 23	<0.04	<0.04	<0.04	0.04±0.04	<0.04	<0.03	0.07±0.04	<0.03	<0.04	<0.04	<0.04
Jan 23 - Jan 30	0.02±0.02	0.03±0.02	<0.02	<0.02	<0.02	0.02±0.02	0.04±0.02	0.05±0.02	0.10±0.02	0.05±0.02	0.10±0.02
Jan 30 - Feb 6	(a)	<0.05	<0.08	<0.05	<0.05	(a)	(a)	(b)	(a)	<0.05	0.17±0.06
Feb 6 - Feb 13	(a)	<0.03	<0.03	<0.02	<0.03	(a)	(a)	(a)	(a)	<0.03	0.05±0.03
Feb 13 - Feb 20	0.02±0.02	0.04	0.04±0.04	<0.03	<0.04	<0.01	0.04±0.02	0.02±0.02	0.03±0.02	<0.04	0.06±0.04
Feb 20 - Feb 27	0.02±0.02	0.02	0.04±0.02	<0.01	0.03±0.02	0.02±0.02	0.02±0.02	<0.02	0.02±0.02	0.04±0.02	0.04±0.02
Feb 27 - Mar 6	<0.03	0.03±0.03	<0.03	0.05±0.02	0.03±0.03	0.02±0.02	0.05±0.04	0.04±0.02	0.06±0.03	0.06±0.03	0.08±0.03
Mar 6 - Mar 13	<0.02	<0.02	<0.02	(b)	<0.02	<0.02	0.02±0.03	<0.02	0.02±0.02	0.04±0.02	0.02±0.02
Mar 13 - Mar 20	0.03±0.03	<0.03	0.06±0.03	<0.03	<0.03	0.07±0.03	0.14±0.04	0.07±0.03	0.06±0.03	0.04±0.03	0.22±0.16 (c)
Mar 20 - Mar 27	0.02±0.02	<0.02	<0.02	<0.02	<0.02	0.02±0.02	<0.02	<0.02	0.08±0.02	0.04±0.02	(b)
Mar 27 - Apr 3	<0.02	0.03±0.02	<0.02	<0.02	<0.02	<0.02	<0.02	0.02±0.02	0.04±0.02	<0.02	0.04±0.02
Apr 3 - Apr 10	<0.03	<0.03	<0.03	<0.03	0.03±0.03	<0.03	<0.03	<0.02	0.06±0.03	0.11±0.03	0.47±0.04
Apr 10 - Apr 17	<0.05	<0.05	<0.05	<0.05	<0.05	<0.05	<0.05	<0.05	<0.05	<0.05	0.08±0.05
Apr 17 - Apr 24	<0.03	<0.03	<0.03	<0.03	<0.03	0.05±0.03	<0.03	0.11±0.03	0.08±0.03	0.07±0.03	0.17±0.03
Apr 24 - May 1	<0.02	<0.03	<0.07 (c)	<0.02	0.04±0.02	0.02±0.02	0.04±0.02	0.02±0.02	0.03±0.02	0.04±0.02	0.09±0.02
May 1 - May 8	0.02±0.02	<0.02	0.02±0.02	<0.02	0.03±0.02	<0.02	0.05±0.02	0.02±0.02	<0.02	<0.02	0.12±0.02
May 8 - May 15	0.04±0.04	0.03±0.04	0.08±0.04	<0.04	0.05±0.04	0.03±0.04	(b)	0.11±0.05	<0.04	(b)	0.04±0.04
May 15 - May 22	<0.02	0.02±0.02	<0.02	<0.02	0.05±0.02	0.08±0.02	0.03±0.02	0.10±0.02	0.04±0.02	0.03±0.02	0.03±0.02
May 22 - May 29	<0.03	<0.03	0.03±0.03	0.02±0.02	<0.03	<0.03	0.14±0.04	0.03±0.03	0.09±0.03	0.04±0.03	0.07±0.04
May 29 - June 5	<0.03	<0.03	<0.03	<0.02	0.03±0.03	0.04±0.03	0.03±0.03	0.06±0.03	<0.03	0.05±0.03	0.07±0.03
June 5 - June 12	<0.03	<0.03	<0.06 (c)	<0.02	<0.03	<0.03	<0.03	<0.03	0.05±0.03	0.03±0.02	<0.03
June 12 - June 19	0.05±0.03	0.04±0.03	0.04±0.04	0.02±0.02	0.06±0.03	0.04±0.03	<0.03	0.22±0.03	0.03±0.03	0.02±0.03	0.03±0.03
June 19 - June 26	<0.03	<0.03	<0.03	<0.03	<0.03	<0.03	<0.03	<0.03	0.03±0.03	<0.02	0.06±0.03

(a) Station inaccessable - no sample until following week.
(b) Instrument out of service.
(c) Instrument malfunction - partial sample.

TABLE A-2

SEMI-ANNUAL SUMMARY OF RADIOACTIVE GASEOUS EFFLUENTS
JANUARY - JUNE 1975

		Jan	Feb	Mar	Apr	May	June	Total
1. Total Noble Gases	(Ci)							
(a) Main Stack		9.87 E+3	5.37 E+3	9.13 E+2	2.30 E+3	3.38 E+3	4.03 E+3	2.59 E+4
(b) Reactor Building Vent		2.06 E+3	1.15 E+3	3.27 E+3	3.97 E+3	4.37 E+3	5.30 E+3	2.01 E+4
2. Total Halogens (a)								
(a) Main Stack		3.35 E-1	3.12 E-2	1.88 E-2	7.95 E-2	4.31 E-2	3.27 E-2	5.40 E-1
(b) Reactor Building Vent		1.09 E-1	5.54 E-2	6.62 E-2	1.26 E-1	1.63 E-1	1.13 E-1	6.33 E-1
3. Total Particulates (a)	(Ci)							
(a) Main Stack		4.01 E-4	4.61 E-4	5.94 E-4	1.04 E-3	1.82 E-3	4.24 E-3	8.56 E-3
(b) Reactor Building Vent		1.10 E-3	1.72 E-3	2.62 E-3	3.37 E-3	3.81 E-3	5.21 E-3	1.78 E-2
4. Total Particulate Gross Alpha Radioactivity	(Ci)							
(a) Main Stack		(b)	(b)	2.16 E-8	(b)	1.51 E-7	(b)	
(b) Reactor Building Vent		(b)	(b)	<6.59 E-8	(b)	<1.75 E-8	(b)	
5. Total Tritium	(Ci)							
(a) Main Stack		5.60 E-1	5.70 E-1	8.30 E-1	5.10 E-1	1.21 E+0	1.34 E+0	5.02 E+0
(b) Reactor Building Vent		3.40 E+0	3.81 E+0	5.91 E+0	5.75 E+0	4.11 E+0	8.53 E+0	3.15 E+1
6. Maximum 24-hr Noble Gas Release	(Date)	1-26	2-17	3-26	4-10	5-3	6-30	
		2.35 E+3	1.15 E+3	5.03 E+2	1.08 E+3	4.57 E+2	5.26 E+2	
7. Percent of Applicable Limit for Noble Gases		2.24	1.37	1.23	1.89	2.12	2.64	1.95
8. Percent of Applicable Limit for Halogens and Particulates		77.46	39.76	42.69	84.23	(c)	12.82	
9. Isotopes Released	(Ci)							
A. Halogens								
Iodine-131		4.44 E-1	8.66 E-2	8.50 E-2	2.06 E-1	2.06 E-1	1.46 E-1	1.17 E+0
Iodine-133		(b)	7.75 E-2	(b)	1.96 E-1	(b)	(b)	
Iodine-135		(b)	1.35 E-1	(b)	2.39 E-1	(b)	(b)	
B. Particulates								
Chromium-51		2.88 E-5	-	-	-	-	-	2.88 E-5
Manganese-54		3.66 E-5	2.19 E-5	1.39 E-5	1.40 E-5	7.51 E-6	5.48 E-6	9.93 E-5
Cobalt-58		6.18 E-5	7.56 E-6	5.06 E-7	1.37 E-6	-	6.15 E-6	7.74 E-5
Iron-59		1.17 E-5	-	-	-	-	-	1.17 E-5
Cobalt-60		6.45 E-5	4.80 E-5	2.89 E-5	3.51 E-5	2.95 E-5	8.57 E-6	1.74 E-4
Zirc/Niob-95		2.24 E-6	1.61 E-7	-	5.84 E-7	-	6.22 E-6	9.21 E-6
Silver-110m		-	-	-	-	-	1.13 E-4	1.13 E-4
Cesium-134		5.79 E-5	2.96 E-5	5.17 E-6	2.47 E-5	3.59 E-5	2.87 E-5	1.94 E-4
Cesium-136		-	2.63 E-6	-	-	-	-	2.63 E-6
Cesium-137		1.54 E-4	1.61 E-4	4.71 E-5	8.99 E-5	1.19 E-4	1.03 E-4	6.73 E-4
Barium-Lanth.-140		1.06 E-3	1.87 E-3	2.99 E-3	4.08 E-3	5.23 E-3	8.96 E-3	2.42 E-2
Cerium-141		2.27 E-5	3.59 E-5	1.31 E-4	1.56 E-4	2.09 E-4	2.19 E-4	7.74 E-4
Cerium-144		-	-	-	-	2.40 E-5	-	2.40 E-5
Strontium-89		(b)	(b)	1.58 E-3	(b)	2.04 E-3	-	-
Strontium-90		(b)	(b)	7.37 E-6	(b)	1.17 E-5	-	-
C. Gases (d)								

(a) With lalf-lives greater than 8 days.
(b) Quarterly analysis - Tech. Specs.
(c) Tech. Spec. change to quarterly limit.
(d) Unable to predict due to Augmented Off-Gas System.

BOSTON EDISON COMPANY
GENERAL OFFICES 800 BOYLSTON STREET
BOSTON, MASSACHUSETTS 02199

March 1, 1976

Director
Region I, Inspection and Enforcement
U. S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

Docket No. 50-293
License No. DPR-35

Semi-Annual Report No. 7
Environmental Radiation Monitoring Program

Gentlemen:

In accordance with Pilgrim Station, Technical Specification 6.9.C.2, we are hereby submitting our seventh Semi-Annual Report. A separate report covering Technical Specification 6.9.C.1 and entitled, "Semiannual Summary of Radioactive Effluents" has been sent under separate cover.

Very truly yours,

Original Signed by

G. Carl Andognini
Manager -
Nuclear Operations

cc: Director
Office of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555 (20)

Director
Office of Management Information
and Program Control
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555 (2)

BOSTON EDISON COMPANY

PILGRIM NUCLEAR GENERATING STATION

Environmental Radiation Monitoring Program

SEMIANNUAL REPORT NO. 7
JULY 1, 1975 THROUGH DECEMBER 31, 1975

Prepared By

Joel I. Cehn
Environmental Sciences Group
Nuclear Engineering Department

March 1976

Approved By: G. James Davis

G. James Davis, Manager
Environmental Sciences Group

I. INTRODUCTION AND SUMMARY

This report discusses the data accumulated by the Environmental Radiation Surveillance Program during the semiannual period July 1 through December 31, 1975.

Pilgrim Station operation during the reporting period is detailed in a separate report entitled "Operating and Maintenance Annual Report, 1975." Power levels during this period were limited, administratively, to 70 percent of full power from July through October, and to 60 percent of full power during November and December. These limits on power levels were set by Boston Edison Company for reasons of operation and maintenance. Plant capacity factors (a measure of electrical output as a percentage of full output 100 percent of the time) during the reporting period were: July, 43 percent; August, 53 percent; September, 19 percent; October, 3 percent; November, 50 percent; and December, 53 percent.

The Environmental Radiation Surveillance Program was implemented during the reporting period as required by the Nuclear Regulatory Commission. Plant-related radioactivity levels in marine life and sediments decreased from 1974 levels, as did liquid effluent releases to Cape Cod Bay. The current levels of Mn-54, Co-60, Zn-65 and Cs-137 in these media are less than 1 picocurie per gram.

Gaseous radioiodine was detected onsite at levels less than 1 picocurie per cubic meter. This I-131 was attributed to releases from the reactor building vent.

TABLE 6

AIR PARTICULATES - GROSS BETA CONCENTRATIONS IN WEEKLY SAMPLES (pCi/m^3)

Period (1975)	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property Line	Overlook Area	Pedestrian Bridge	East Breakwater	Rocky Hill Road (East)	Warehouse
June 26 - July 2	0.09	0.07	0.07 [a]	0.07	0.06	0.07	0.05	0.07	0.06	0.06	0.07
July 2 - July 10	0.04	0.05	0.06 [e]	0.06	0.05	0.05	0.04	0.05	0.05	0.04	0.05
July 10 - July 17	0.04	0.04	0.04 [a]	0.04	0.03	0.03	0.02	0.05	0.03	0.03	0.03
July 17 - July 24	0.04	0.04	0.04 [a]	0.04	0.05	0.02	0.03	0.04	0.04	0.04	0.04
July 24 - July 31	0.04	0.04	0.05	0.04	0.05	0.06	0.04	0.02 [a]	0.03	0.05	0.04
July 31 - Aug 8	0.04	0.05	0.04	0.02	0.05	0.04	0.03	0.05	0.05	0.05	0.02 [a]
Aug 8 - Aug 14	0.07	0.07	0.07	0.07	0.06	0.07	0.06	0.06	0.06	0.01 [a]	0.06
Aug 14 - Aug 21	0.04	0.05	0.05	0.04	0.04	0.02	0.0[illegible]	0.05	0.04	0.05 [a]	0.04
Aug 21 - Aug 28	0.04	0.04	0.05	0.05	0.04	0.05	0.03	0.05	0.05	0.05	0.06
Aug 28 - Sep 4	0.04	[illegible]	0.04	0.04	0.03	(b)	0.04	0.04	0.05	0.04	0.05
Sep 4 - Sep 11	0.04	0.05	0.05	0.05	0.04	0.04 [c]	0.03 [a]	0.05	0.05	0.04	0.05
Sep 11 - Sep 18	0.03	0.03	0.03	0.04	0.04	0.04	0.04	0.04	0.04 [a]	0.03	0.05
Sep 18 - Sep 25	0.03	0.03	0.03	0.03	0.04	0.03	0.03 [a]	0.03	(e)	0.04	0.03 [a]
Sep 25 - Oct 2	0.03	0.03	0.03	0.03	0.03	0.03	0.04	0.03	(e)	0.03 [a]	(e)
Oct 2 - Oct 9	0.03	(e)	0.04	0.04	0.04	0.05	0.04	0.04	0.04	0.04	0.03
Oct 9 - Oct 16	0.04	0.04	0.05	0.04	0.02	0.04	0.03	0.04	0.03	0.04	0.04
Oct 16 - Oct 23	(e)	0.03	0.03	0.04	0.02	0.03	0.02	0.04	0.04	0.03	<0.01
Oct 23 - Oct 30	0.02	0.04	0.01	0.03	0.03	0.03	0.03	0.02 [d]	0.03	0.03	0.02
Oct 30 - Nov 6	0.03	0.03	0.04	0.04	0.03	0.03	0.02	0.03	0.03	0.04	0.02
Nov 6 - Nov 13	0.03	0.03	0.03	(e)	0.04	0.03	0.03	0.03	0.03	0.02	0.03
Nov 13 - Nov 20	0.04	0.04	0.05	0.04	0.04	0.04	0.03	0.04	0.04	0.04	0.03
Nov 20 - Nov 26	0.03	0.03	0.04	0.03	0.04	0.02	0.07	0.04	0.03	0.04	0.02
Nov 26 - Dec 4	0.03	0.04	0.04	0.04	0.04	0.03	0.04	0.05	0.06	0.04	0.05
Dec 4 - Dec 11	0.03	0.03	0.03	0.04	0.03	0.03	0.08	0.03	0.03	0.04	0.05 [a]
Dec 11 - Dec 17	0.03	0.02	0.03	0.04	0.03	0.03	0.03	0.03	0.04	0.04	0.10
Dec 17 - Dec 23	(b)	0.03	0.02	0.03	0.04	0.02	0.04	0.03	0.04	0.04	0.04
Dec 23 - Dec 30	0.03 [c]	0.03	0.04	0.04	0.03	0.04	0.02	0.03	0.04	0.04	0.06

(a) Instrument malfunction - incomplete or low volume sample.
(b) Instrument inaccessable - no sample.
(c) Two-week sample.
(d) Sample removed 10/28.
(e) Instrument malfunction - no sample.

TABLE 7

AIR PARTICULATES - GROSS GAMMA CONCENTRATION IN MONTHLY COMPOSITES
($cpm/m^3 \times 10^{-3}$) (a)

Collection Period (1975)	Control	Offsite			Onsite						
	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property Line	Overlook Area	Pedestrian Bridge	East Breakwater	Rocky Hill Road (East)	Warehouse
July	9	8	9	13	6	3	3	9	9	5	4
August	4	7	<3	9	8	<3	6	7	7	11	7
September	5	<3	5	7	9	4	<3	<3	14	8	<3
October	11	9	7	8	4	5	20	7	4	<3	6
November	4	<3	<3	5	<3	6	8	<3	<3	<3	8
December	<3	<3	12	<3	<3	<3	<3	<3	3	<3	10

(a) Analytical error is ±2 or 10%, whichever is larger.

TABLE 7A

AIR PARTICULATES - GAMMA ISCTOPE CONCENTRATION IN MONTHLY COMPOSITES
(pCi/m^3) (a)

Collection Period (1975)	Control	Offsite			Onsite						
	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property Line	Overlook Area	Pedestrian Bridge	East Breakwater	Rocky Hill Road (East)	Warehouse
July											
Cs-137	<0.006	<0.005	<0.009	<0.007	<0.006	<0.007	<0.006	<0.007	<0.006	<0.005	<0.005
Mn-54	<0.007	<0.005	<0.008	<0.007	<0.005	<0.006	<0.006	<0.007	<0.005	<0.005	<0.005
Zn-65	<0.01	<0.01	<0.01	<0.02	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.009
Co-58	<0.006	<0.004	<0.008	<0.006	<0.005	<0.006	<0.005	<0.007	<0.005	<0.005	<0.004
Co-60	<0.006	<0.005	<0.01	<0.008	<0.005	<0.006	<0.007	<0.006	<0.005	<0.005	<0.005
Other				0.07±0.07 K-40			0.06±0.05 K-40	0.3 ±0.08 K-40 0.009 ±0.007 Th-228	0.07±0.05 K-40	0.06±0.05 K-40	
October											
Cs-137	<0.009	<0.008	<0.007	<0.007	<0.006	<0.007	<0.007	<0.007	<0.006	<0.008	<0.007
Mn-54	<0.008	<0.008	<0.006	<0.007	<0.006	<0.006	<0.007	<0.008	<0.006	<0.008	<0.007
Zn-65	<0.02	<0.02	<0.01	<0.01	<0.01	<0.01	<0.02	<0.01	<0.01	<0.02	<0.02
Co-58	<0.008	<0.008	<0.006	<0.007	<0.007	<0.006	<0.007	<0.01	<0.005	<0.008	<0.007
Co-60	<0.007	<0.009	<0.006	<0.007	<0.007	<0.006	<0.007	<0.008	<0.006	<0.009	<0.007
Other	0.26±0.09 K-40				0.1 ±0.09 K-40	0.09±0.09 K-40	0.02±0.01 K-40	0.15±0.08 K-40	0.006±0.004 Th-228		

(a) Results of Ge(Li) spectrometry. Analysis required quarterly.

TABLE 7B

AIR PARTICULATES - STRONTIUM-90 CONCENTRATION IN QUARTERLY COMPOSITES
($pCi/m^3 \times 10^{-4}$)

Collection Period (1975)	Control	Offsite			Onsite						
	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property Line	Overlook Area	Pedestrian Bridge	East Breakwater	Rocky Hill Road (East)	Warehouse
Jan - Mar	13 $\pm$ 3	13 $\pm$ 2	<2	<2	12 $\pm$ 2	8 $\pm$ 2	11 $\pm$ 4	12 $\pm$ 2	11 $\pm$ 2	14 $\pm$ 4	13 $\pm$ 2
Apr - June	<6	19 $\pm$ 3	17 $\pm$ 3	<4	6 $\pm$ 2	21 $\pm$ 5	14 $\pm$ 4	19 $\pm$ 4	16 $\pm$ 2	10 $\pm$ 2	17 $\pm$ 4
July - Sep	6 $\pm$ 1	5 $\pm$ 1	6 $\pm$ 1	5 $\pm$ 1	7 $\pm$ 2	4 $\pm$ 1	<3	6 $\pm$ 1	4 $\pm$ 1	5 $\pm$ 1	6 $\pm$ 1
Oct - Dec	3 $\pm$ 1	4 $\pm$ 1	3 $\pm$ 1	5 $\pm$ 1	3 $\pm$ 1	2 $\pm$ 2	3 $\pm$ 2	1 $\pm$ 1	7 $\pm$ 2	3 $\pm$ 1	<3

TABLE 8

PARTICULATE IODINE-131 IN AIR SAMPLES (pCi/m^3)
JULY - DECEMBER 1975

	Control	Offsite			Onsite						
Collection Period	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property Line	Overlook Area	Pedestrian Bridge	East Breakwater	Rocky Hill Road (East)	Warehouse
June 26 - July 2	<0.02	<0.03	<0.04 (a)	<0.03	<0.03	<0.03	<0.03	<0.03	<0.03	<0.03	<0.03
July 2 - July 10	<0.02	<0.02	<0.05 (a)	<0.02	<0.02	<0.02	<0.02	<0.03	<0.02	<0.02	<0.02
July 10 - July 17	<0.04	<0.04	<0.09 (a)	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04
July 17 - July 24	<0.05	<0.05	<0.11 (a)	<0.05	<0.05	<0.06	<0.05	<0.05	<0.05	<0.05	<0.05
July 24 - July 31	<0.03	<0.03	<0.03	<0.03	0.03±0.02	<0.03	<0.03	<0.04 (f)	<0.03	<0.03	<0.03
July 31 - Aug 8	<0.03	<0.03	<0.03	<0.03	<0.03	<0.03	<0.03	<0.03	<0.03	<0.03	<0.05 (a)
Aug 8 - Aug 14	<0.02	<0.02	<0.02	<0.02	<0.02	<0.03	<0.02	<0.03	<0.02	<0.02 (a)	<0.02
Aug 14 - Aug 21	<0.02	<0.02	<0.02	0.02±0.01	0.02±0.01	<0.02	<0.02	<0.02	<0.02	0.02±0.01 (a)	<0.02
Aug 21 - Aug 28	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02
Aug 28 - Sep 4	<0.05	<0.05	<0.05	<0.05	<0.05	(b)	<0.05	<0.05	<0.05	<0.05	<0.05
Sep 4 - Sep 11	<0.04	<0.04	<0.03	<0.04	<0.03	<0.03 (c)	<0.05 (a)	<0.04	<0.04	<0.04	<0.04
Sep 11 - Sep 18	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04 (a)	<0.04	<0.04
Sep 18 - Sep 25	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.06 (a)	<0.04	(e)	<0.04	<0.05
Sep 25 - Oct 2	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	(e)	<0.03 (a)	(e)
Oct 2 - Oct 9	<0.02	(e)	<0.02	<0.02	<0.02	<0.02	<0.02	<0.03	<0.02	<0.02	<0.02
Oct 9 - Oct 16	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	0.02±0.01	<0.02
Oct 16 - Oct 23	(e)	<0.02	<0.02	<0.02	<0.02	0.02±0.02	0.04±0.02	<0.02	0.03±0.02	<0.02	0.03±0.02
Oct 23 - Oct 30	<0.02	<0.02	0.02±0.01	<0.02	<0.02	<0.02	<0.02	<0.03 (d)	<0.02	<0.02	<0.03
Oct 30 - Nov 6	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.03	<0.02	<0.02	<0.02	<0.03
Nov 6 - Nov 13	<0.02	0.02±0.01	<0.02	(e)	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02
Nov 13 - Nov 20	<0.03	<0.04	<0.04	<0.04	<0.03	<0.03	<0.04	<0.04	<0.04	<0.03	<0.03
Nov 20 - Nov 26	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.03	<0.02	<0.02	<0.02
Nov 26 - Dec 4	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02
Dec 4 - Dec 11	<0.02	<0.03	<0.02	<0.03	<0.02	<0.02	<0.03	<0.03	<0.02	<0.02	<0.03 (a)
Dec 11 - Dec 17	<0.03	<0.03	<0.03	<0.03	<0.03	<0.03	<0.03	<0.03	<0.03	<0.03	<0.04
Dec 17 - Dec 23	(b)	<0.05	<0.04	<0.04	<0.04	<0.04	<0.04	<0.05	<0.05	0.06±0.03	<0.05
Dec 23 - Dec 30	<0.01 (c)	0.02±0.01	<0.03	0.02±0.01	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	0.02±0.02

(a) Instrument malfunction - incomplete or low volume sample.
(b) Instrument inaccessable - no sample.
(c) Two week sample.
(d) Sample removed 10/28.
(e) Instrument malfunction - no sample.
(f) Counting errors are ±2 Standard Deviations: "less than" values are 3 Standard Deviations.

TABLE 8A

GASEOUS IODINE-131 IN AIR SAMPLES (pCi/m^3)
JULY - DECEMBER 1975

	Control	Offsite			Onsite						
Collection Period	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property Line	Overlook Area	Pedestrian Bridge	East Breakwater	Rocky Hill Road (East)	Warehouse
June 26 - July 2	<0.03	<0.04	<0.06 [a]	<0.04	<0.03	<0.04	<0.04	<0.04	<0.03	<0.03	<0.04
July 2 - July 10	<0.05	<0.05	<0.11 [a]	<0.05	<0.05	<0.05	<0.05	0.06±0.03	<0.05	0.05±0.03	<0.05
July 10 - July 17	<0.05	<0.05	<0.11 [a]	<0.05	<0.04	<0.05	<0.05	0.09±0.04	<0.05	<0.05	<0.05
July 17 - July 24	<0.04	<0.05	<0.10 [a]	<0.05	<0.04	<0.05	<0.05	0.08±0.04	0.06±0.03	<0.04	<0.05
July 24 - July 31	<0.03	<0.03	<0.03	<0.03	0.05±0.02	<0.03	<0.03	0.06±0.03 [a]	0.04±0.02	<0.04	<0.03
July 31 - Aug 8	<0.04	<0.04	<0.04	<0.05	<0.04	<0.04	<0.05	<0.04	<0.04	<0.05	0.24±0.06 [a]
Aug 8 - Aug 14	<0.03	<0.04	0.07±0.02	<0.03	<0.03	<0.04	<0.03	<0.04	<0.04	<0.03 [a]	0.11±0.02
Aug 14 - Aug 21	<0.04	<0.04	0.05±0.03	<0.04	<0.04	<0.04	<0.04	<0.04	0.06±0.03	<0.04 [a]	0.06±0.03
Aug 21 - Aug 28	<0.03	<0.03	<0.03	<0.03	<0.03	<0.03	<0.03	0.06±0.03	0.08±0.02	<0.04	0.08±0.02
Aug 28 - Sep 4	<0.04	<0.04	<0.04	<0.04	<0.04	(b)	<0.04	0.07±0.03	0.09±0.03	0.07±0.03	0.18±0.03
Sep 4 - Sep 11	<0.05	<0.06	<0.06	<0.06	<0.05	<0.04 [c]	<0.08 [a]	<0.06	0.10±0.04	<0.06	<0.06
Sep 11 - Sep 18	<0.07	<0.07	<0.08	<0.07	<0.07	<0.08	<0.08	<0.08	<0.08 [a]	0.10±0.05	0.58±0.06
Sep 18 - Sep 25	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.07 [a]	<0.04	(e)	<0.04	<0.05 [a]
Sep 25 - Oct 2	<0.03	<0.03	<0.03	<0.03	<0.03	<0.03	<0.03	<0.03	(e)	<0.04 [e]	(e)
Oct 2 - Oct 9	<0.04	(e)	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.03	<0.04	0.05±0.02
Oct 9 - Oct 16	<0.04	<0.03	<0.04	<0.04	<0.04	<0.04	<0.05	<0.04	<0.04	<0.04	<0.05
Oct 16 - Oct 23	(e)	<0.04	<0.04	<0.04	<0.04	<0.04	0.04±0.03	<0.04	0.04±0.03	<0.04	0.04±0.03
Oct 23 - Oct 30	<0.03	<0.03	<0.03	<0.03	<0.03	<0.03	<0.03	<0.04 [d]	<0.03	<0.03	<0.04
Oct 30 - Nov 6	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	0.05±0.03	<0.03	<0.04	<0.04	<0.04
Nov 6 - Nov 13	<0.04	<0.04	<0.04	(e)	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04
Nov 13 - Nov 20	<0.03	<0.03	0.03±0.02	<0.03	<0.03	0.03±0.02	<0.03	0.03±0.02	<0.03	<0.03	0.03±0.02
Nov 20 - Nov 26	<0.05	<0.05	<0.05	<0.05	<0.05	<0.05	0.07±0.04	<0.06	<0.05	<0.05	<0.05
Nov 26 - Dec 4	<0.03	<0.03	<0.03	<0.03	<0.02	<0.02	<0.03	<0.03	0.04±0.02	<0.03	0.04±0.02
Dec 4 - Dec 11	<0.05	<0.05	<0.05	<0.05	<0.05	<0.05	0.22±0.04	<0.05	<0.05	<0.05	<0.06 [a]
Dec 11 - Dec 17	<0.03	<0.04	<0.03	<0.04	<0.03	<0.03	<0.03	<0.04	<0.03	<0.03	0.17±0.03
Dec 17 - Dec 23	(b)	<0.06	<0.06	<0.06	<0.06	<0.06	0.06±0.04	<0.07	<0.06	0.10±0.05	<0.07
Dec 23 - Dec 30	<0.02 [c]	<0.03	<0.04	<0.03	<0.03	<0.03	<0.03	<0.03	0.04±0.02	<0.02	0.04±0.02

(a) Instrument malfunction - incomplete or low volume sample.
(b) Instrument inaccessable - no sample.
(c) Two week sample.
(d) Sample removed 10/28.
(e) Instrument malfunction - no sample.
(f) Counting errors are ±2 Standard Deviations; "less than" values are 3 Standard Deviations.

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT (1975)

GASEOUS EFFLUENTS–ELEVATED RELEASE

Nuclides Released	Unit	CONTINUOUS MODE Quarter	CONTINUOUS MODE Quarter	BATCH MODE Quarter	BATCH MODE Quarter

1. Fission gases

Nuclides Released	Unit	Continuous Quarter	Continuous Quarter	Batch Quarter	Batch Quarter
krypton-85	Ci	. E	. E	. E	. E
krypton-85m	Ci	. E	. E	. E	. E
krypton-87	Ci	. E	. E	. E	. E
krypton-88	Ci	. E	. E	. E	. E
xenon-133	Ci	. E	. E	. E	. E
xenon-135	Ci	. E	. E	. E	. E
xenon-135m	Ci	. E	. E	. E	. E
xenon-138	Ci	. E	. E	. E	. E
Others (specify)	Ci	. E	. E	. E	. E
	Ci	. E	. E	. E	. E
	Ci	. E	. E	. E	. E
unidentified	Ci	1. 74E+4	130 E+4	. E	. E
Total for period	Ci	1. 74E+4	1. 30E+4	. E	. E

2. Iodines

Nuclides Released	Unit	Continuous Quarter	Continuous Quarter	Batch Quarter	Batch Quarter	
iodine-131	Ci	4.40 E-1	1.40 E-2	. E	. E	4.54E-1
iodine-133	Ci	5.19 E-1	2.53 E-2	. E	. E	5.44E-1
iodine-135	Ci	5.74 E-1	4.44 E-2	. E	. E	6.18E-1
Total for period	Ci	1.53 E+0	2.07 E-1	. E	. E	

3. Particulates

Nuclides Released	Unit	Continuous Quarter	Continuous Quarter	Batch Quarter	Batch Quarter
strontium-89	Ci	9.12 E-3	6.19 E-3	. E	. E
strontium-90	Ci	5.35 E-5	3.54 E-5	. E	. E
cesium-134	Ci	1.75 E-5	6.80 E-6	. E	. E
cesium-137	Ci	2.13 E-4	1.29 E-4	. E	. E
barium-lanthanum-140	Ci	2.00 E-2	1.40 E-2	. E	. E
manganese-54	Ci	1.86 E-5	8.46 E-6	. E	. E
cobalt-58	Ci	6.12 E-7	<MDA	. E	. E
zinc-65	Ci	3.47 E-6	<MDA	. E	. E
cobalt-60	Ci	3.60 E-5	3.81 E-5	. E	. E
zirconium-niobium-95	Ci	3.08 E-5	8.85 E-7		
cerium-141	Ci	2.64 E-5	2.21 E-6		
cerium-144	Ci	3.27 E-6	<MDA		
nepunium-239	Ci	1.39 E-5	<MDA		
unidentified	Ci				
Total for period	Ci	2.95 E-2	2.04 E-2		

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT (1975)

GASEOUS EFFLUENTS–GROUND-LEVEL RELEASES

		CONTINUOUS MODE		BATCH MODE	
Nuclides Released	Unit	Quarter	Quarter	Quarter	Quarter

1. Fission gases

krypton-85	Ci	. E	. E	. E	. E
krypton-85m	Ci	. E	. E	. E	. E
krypton-87	Ci	. E	. E	. E	. E
krypton-88	Ci	. E	. E	. E	. E
xenon-133	Ci	. E	. E	. E	. E
xenon-135	Ci	. E	. E	. E	. E
xenon-135m	Ci	. E	. E	. E	. E
xenon-138	Ci	. E	. E	. E	. E
Others (specify)	Ci	. E	. E	. E	. E
	Ci	. E	. E	. E	. E
	Ci	. E	. E	. E	. E
unidentified	Ci	1. 55E 4	1. 33E 4	. E	. E
Total for period	Ci	1. 55E 4	1. 33E 4	. E	. E

2. Iodines

iodine-131	Ci	4. 92E-1	7. 35E-2	. E	. F	5.66E-1
iodine-133	Ci	6. 55E-1	2. 42E-1	. E	. E	8.97E-1
iodine-135	Ci	8. 09E-1	4. 36E-1	. E	. E	1.33E0
Total for period	Ci	2. 04E+0	1. 72E+0	. E	. E	

3. Particulates

strontium-89	Ci	6,57 E-3	5.62 E-3	. E	. E
strontium-90	Ci	3.12 E-5	2.99 E-5	. E	. E
cesium-134	Ci	1.36 E-4	7.16 E-5	. E	. E
cesium-137	Ci	4.92 E-4	5.05 E-4	. E	. E
barium-lanthanum-140	Ci	4.92 E-1	7.35 E-2	. E	. E
chromium-51	Ci	7.55 E-5	<MDA	. E	. E
manganese-54	Ci	1.81 E-4	1.02 E-4	. E	. E
cobalt-58	Ci	2.11 E-6	7.52 E-6	. E	. E
iron-59	Ci	<MDA	1.73 E-6	. E	. E
zinc-65	Ci	<MDA	3.36E-5		
cobalt-60	Ci	2.13E-4	4.65E-4		
zirconium-niobium-95	Ci	5.80E-5	8.55F-5		
silver-110m	Ci	2.18E-3	1.44E-3		
cerium-141	Ci	4.13E-4	1.20E-4		
cerium-144	Ci	<MDA	4.03E-6		
Total for period	Ci	5.02E-1	8.70E-2		

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT (1975)

GASEOUS EFFLUENTS–SUMMATION OF ALL RELEASES

	Unit	Quarter 3	Quarter 4	Est. Total Error, %
A. Fission & activation gases				
1. Total release	Ci	3.29 E+4	2.63 E+4	6.50 E+1
2. Average release rate for period	μCi/sec	4.14 E+3	2.31 E+3	
3. Percent of Technical specification limit	%	2.52 E+0	2.32 E+0	
B. Iodines				
1. Total iodine-131	Ci	9.32 E-1	8.75 E-2	6.00 E+1
2. Average release rate for period	μCi/sec	1.17 E-1	1.10 E-2	
3. Percent of technical specification limit	%	1.57 E+1	2.20 E+0	
C. Particulates				
1. Particulates with half-lives >8 days	Ci	5.32 E-1	1.02 E-1	6.50 E1
2. Average release rate for period	μCi/sec	6.69 E-2	1.28 E-2	
3. Percent of technical specification limit	%	1.49 E+1	2.46 E+0	
4. Gross alpha radioactivity	Ci	5.78 E-6	8.27 E-7	
D. Tritium				
1. Total release	Ci	2.48 E 1	1.31 E 1	5.00 E+1
2. Average release rate for period	μCi/sec	3.12 E 0	1.65 E 0	
3. Percent of technical specification limit	%	. E	. E	

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT (1975)

GASEOUS EFFLUENTS–SUMMATION OF ALL RELEASES

	Unit	Quarter 3	Quarter 4	Est. Total Error, %
A. Fission & activation gases				
1. Total release	Ci	3.29 E+4	2.63 E+4	6.50 E+1
2. Average release rate for period	μCi/sec	4.14 E+3	2.31 E+3	
3. Percent of Technical specification limit	%	2.52 E+0	2.32 E+0	
B. Iodines				
1. Total iodine-131	Ci	9.32 E-1	8.75 E-2	6.00 E+1
2. Average release rate for period	μCi/sec	1.17 E-1	1.10 E-2	
3. Percent of technical specification limit	%	1.57 E+0	2.20 E0	
C. Particulates				
1. Particulates with half-lives >8 days	Ci	5.32 E-1	1.02 E-1	6.50 E1
2. Average release rate for period	μCi/sec	6.69 E-2	1.28 E-2	
3. Percent of technical specification limit	%	1.49 E+1	2.48 E+0	
4. Gross alpha radioactivity	Ci	5.78 E-6	8.27 E-7	
D. Tritium				
1. Total release	Ci	2.48 E 1	1.31 E 1	5.00 E+1
2. Average release rate for period	μCi/sec	3.12 E 0	1.65 E 0	
3. Percent of technical specification limit	%	. E	. E	

BOSTON EDISON COMPANY

PILGRIM NUCLEAR GENERATING STATION

Environmental Radiation Monitoring Program

SEMIANNUAL REPORT NO. 8
JANUARY 1, 1976 THROUGH JUNE 30, 1976

Prepared By

Joel I. Cehn
Environmental Sciences Group
Nuclear Engineering Department

August 1976

Approved By: G. James Davis

G. James Davis, Manager
Environmental Sciences Group

I. INTRODUCTION AND SUMMARY

This report discusses the data accumulated by the Environmental Radiation Monitoring Program during the period January 1, through June 30, 1976.

Pilgrim Station operated at 19% of capacity during January and 53% of capacity during June. The unit was shut down for refueling and maintenance from January 29 to June 4.

The Environmental Radiation Monitoring Program was implemented during the reporting period as required by the Nuclear Regulatory Commission. Plant-related radioactivity levels in marine life and sediments increased from 1975 levels, as did liquid effluent releases to Cape Cod Bay. The predominant plant-related radionuclides detected in these media were Cs-137, Co-60 and Mn-54. Activity levels of these nuclides did not exceed 1 picocurie per gram of sample. The dose equivalent associated with the maximum activity concentrations detected was calculated to be less than 1 millirem per year for the seafood ingestion pathway.

The only plant-related activity detected in terrestrial media was gaseous radioiodine. This was detected onsite at levels less than 1 picocurie per cubic meter. This radioiodine was attributed to short-term releases which followed the uncovering of the reactor for refueling.

TABLE 6

AIR PARTICULATES - GROSS BETA CONCENTRATIONS IN WEEKLY SAMPLES (pCi/m^3)

Period (1976)	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property Line	Overlook Area	Pedestrian Bridge	East Breakwater	Rocky Hill Road (East)	Warehouse
Dec 30 - Jan 7	0.02	0.02	0.02	0.03	0.03	0.02	0.02	0.03	0.04	0.04	0.05
Jan 7 - Jan 14	0.03	0.03	0.04	0.03	0.03	0.03	0.02	0.02	0.02	0.03	0.03
Jan 14 - Jan 22	0.05	0.05	0.05	0.04	0.05	(a)	0.04	0.04	(a)	0.05	0.04
Jan 22 - Jan 29	0.04	0.04	0.03	0.05	0.04	0.03(b)	0.03	0.05	0.03(b)	0.04	0.03
Jan 29 - Feb 5	0.03	0.05	0.04	0.05	0.05	0.03	0.03	0.04	0.04	0.04	0.03
Feb 5 - Feb 12	0.04	0.04	0.03	0.03	0.03	0.03	0.03	0.03	0.05	0.04	0.03
Feb 12 - Feb 19	0.04	0.05	0.04	0.05	0.05	0.05	0.04	0.05	0.05	0.05	0.05(c)
Feb 19 - Feb 26	0.05	0.04	0.04	0.03	0.04	0.03	0.04	0.04	0.05	0.04	0.04
Feb 26 - Mar 4	0.03	0.06	0.02	0.02	0.04	0.03	0.04	0.03	0.02	0.03	0.04
Mar 4 - Mar 11	0.02	0.02	0.05(c)	0.03	0.02	0.03	0.03	0.03	0.05	0.02	0.04
Mar 11 - Mar 18	0.01	0.02	(d)	0.03	0.02	0.02	0.02	0.02	0.02	0.03	0.02
Mar 18 - Mar 25	0.02	0.03	0.03	0.03	0.03	0.03	0.03	0.05	0.03	0.03	0.04
Mar 25 - Apr 1	0.02	0.02	0.02	0.0[illegible]	0.02	0.02	0.02	0.02	0.02	0.03	0.04
Apr 1 - Apr 8	0.03	0.03	0.02	0.03	0.03	0.03	0.03	0.03	0.04	0.03	0.03
Apr 8 - Apr 15	0.03	0.03	0.04	0.05	0.03	(d)	0.03	0.03	0.05	0.03	0.04
Apr 15 - Apr 22	0.04	0.05	0.04	0.04	0.04	0.04	0.04	0.04	0.04	0.04	0.04
Apr 22 - Apr 29	0.02	0.02	0.02	0.01	0.02	0.02	0.02	0.02	0.02	0.01	0.02
Apr 29 - May 6	0.03	0.03	0.03	0.02	0.03	0.02	0.02	0.02	0.03	0.02	0.03
May 6 - May 13	0.03	0.03	0.03	0.03	0.03	0.03	0.03	0.03	0.03	0.03	0.03
May 13 - May 20	0.02	0.02	0.02	0.01	0.02	0.02	0.02	0.02(c)	0.01	0.01	0.01
May 20 - May 27	<0.01	0.01	<0.01	<0.01	0.01	<0.01	<0.01	0.01	0.01	0.01	0.02
May 27 - June 3	0.02	0.02	0.02	0.02	0.03	0.02	0.02	0.02	0.01	0.02	0.02
June 3 - June 10	0.03	0.03	0.02	0.02	0.02	0.02	0.02	0.03	0.03	0.02	0.03
June 10 - June 16	0.03	0.03	0.02	0.03	0.04	(a)	<0.01(c)	0.03	0.03	0.05(c)	0.03
June 16 - June 24	0.02	0.02	0.01	0.01	0.02	0.02(b)	0.01	0.01	0.01	0.02	0.02
June 24 - July 1	0.03	0.02	0.02	0.03	0.03	0.03	0.[illegible]	0.03	0.02	0.03	0.03

(a) Instrument station inaccessible
(b) Two week sample
(c) Not full week sample
(d) Instrument malfunction

TABLE 7

AIR PARTICULATES - GROSS GAMMA CONCENTRATION IN MONTHLY COMPOSITES
(cpm/m^3 x 10^{-3}) (a)

	Control	Offsite			Onsite						
Collection Period (1976)	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property Line	Overlook Area	Pedestrian Bridge	East Breakwater	Rocky Hill Road (East)	Warehouse
January	<3	<3	<3	<3	<3	<3	<3	<3	<3	<3	<3
February	10	6	11	5	11	11	4	4	11	8	14
March	<3	6	<3	3	<3	<3	10	9	6	15	4
April	<3	<3	<3	<3	<3	<3	<3	<3	<3	<3	<3
May	4	<3	10	4	6	3	<3	<3	4	<3	7
June	4	8	6	5	7	<3	<3	7	5	7	7

(a) Typical analytical error is $\pm$ 3 x 10^{-3} cpm/m^3
In the counter used, 1 cpm corresponds to 2.2 pCi of Cs-137.

TABLE 7A

AIR PARTICULATES - GAMMA ISOTOPE CONCENTRATION IN MONTHLY COMPOSITES
(pCi/m^3) (a)

	Control	Offsite			Onsite						
Collection Period (1976)	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property Line	Overlook Area	Pedestrian Bridge	East Breakwater	Rocky Hill Road (East)	Warehouse
January											
Cs-137	<.008	<.008	<.008	<.007	<.007	<.008	<.008	<.007	<.008	<.006	<.006
Mn-54	<.008	<.008	<.007	<.007	<.008	<.009	<.009	<.007	<.009	<.007	<.007
Co-58	<.01	<.01	<.01	<.009	<.01	<.01	<.01	<.009	<.01	<.009	<.009
Co-60	<.008	<.009	<.009	<.007	<.008	<.008	<.008	<.007	<.009	<.007	<.007
Zn-65	<.02	<.02	<.02	<.02	<.02	<.02	<.02	<.02	<.02	<.01	<.02
K-40	<.09	<.1	<.1	<.1	.08±.05	<.07	<.2	<.07	.22±.09	<.1	<.06
Be-7	.12±.06	.08±.06	<.1	.08±.07	.15±.06	.14±.06	<.1	<.1	<.1	.16±.08	.06±.05
April											
Cs-137	<.006	<.008	<.008	<.007	<.008	<.01	<.008	<.007	<.008	<.008	<.008
Mn-54	<.006	<.009	<.009	<.006	<.008	<.01	<.008	<.007	<.008	<.008	<.008
Co-58	<.008	<.01	<.01	<.009	<.01	<.01	<.01	<.009	<.009	<.009	<.01
Co-60	<.007	<.008	<.009	<.007	<.008	<.01	<.008	<.007	<.008	<.007	<.008
Zn-65	<.02	<.02	<.02	<.02	<.02	<.02	<.02	<.01	<.02	<.02	<.02
K-40	<.1	<.1	<.1	<.09	<.1	<.02	.21±.09	.07±.06	<.1	<.1	<.1
Be-7	<.08	<.1	.30±.009	.08±.08	<.1	.2±.1	<.1	.10±.08	.1±.1	.20±.09	.10±.10
Others			.013+.009 Th-228								

(a) Results of Ge(Li) spectrometry. Analysis required quarterly

TABLE 8

PARTICULATE IODINE-131 IN AIR SAMPLES (pCi/m^3)

JANUARY - JUNE 1976

	Control	Offsite			Onsite						
Collection Period	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property Line	Overlook Area	Pedestrian Bridge	East Breakwater	Rocky Hill Road (East)	Warehouse
Dec 30 - Jan 7	<0.02	<0.02	<0.03	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02
Jan 7 - Jan 14	<0.04	<0.04	<0.06	<0.04	<0.04	<0.05	<0.05	<0.05	<0.05	<0.04	<0.04
Jan 14 - Jan 22	<0.03	<0.03	<0.03	<0.03	<0.03	(a)	<0.03	<0.03	(a)	<0.03	<0.03
Jan 22 - Jan 29	<0.02	<0.02	<0.02	<0.02	<0.02	<0.01(b)	<0.02	<0.02	<0.01(b)	<0.02	<0.02
Jan 29 - Feb 5	<0.01	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02
Feb 5 - Feb 12	<0.04	<0.04	<0.03	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04
Feb 12 - Feb 19	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.03(c)
Feb 19 - Feb 26	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02
Feb 26 - Mar 4	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02
Mar 4 - Mar 11	<0.04	<0.04	<0.15(c)	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04
Mar 11 - Mar 18	<0.04	<0.04	(d)	<0.04	<0.04	<0.04	<0.05(c)	<0.04	<0.04	<0.04	<0.04
Mar 18 - Mar 25	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02
Mar 25 - Apr 1	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02
Apr 1 - Apr 8	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02
Apr 8 - Apr 15	<0.02	<0.02	<0.02	<0.02	<0.02	(d)	<0.02	<0.02	<0.02	<0.02	<0.02
Apr 15 - Apr 22	<0.02	<0.02	<0.02	<0.02	0.03±0.02	<0.02	<0.02	<0.02	<0.03	<0.02	<0.02
Apr 22 - Apr 29	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02
Apr 29 - May 6	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02
May 6 - May 13	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02
May 13 - May 20	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.04(c)	<0.02	<0.02	<0.02
May 20 - May 27	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.05	<0.04	<0.04	<0.04	<0.04
May 27 - June 3	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02
June 3 - June 10	<0.02	<0.02	<0.02	<0.02	<0.02	<0.02	<0.03	<0.02	<0.02	<0.02	<0.02
June 10 - June 16	<0.03	<0.03	<0.03	<0.03	<0.03	(a)	<0.04	<0.03	<0.03	<0.09(c)	<0.03
June 16 - June 24	<0.03	<0.03	<0.03	<0.03	<0.03	<0.02(b)	<0.03	<0.03	<0.03	<0.03	<0.03
June 24 - July 1	<0.03	<0.03	<0.03	<0.03	<0.03	<0.03	<0.03	<0.03	<0.03	<0.03	<0.03

(a) Instrument Station inaccesible
(b) Two week sample
(c) Not full week sample
(d) Instrument malfunction

TABLE 8A

GASEOUS IODINE-131 IN AIR SAMPLES (pCi/m^3)

JANUARY - JUNE 1976

	Control	Offsite			Onsite						
Collection Period	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property Line	Overlook Area	Pedestrian Bridge	East Breakwater	Rocky Hill Road (East)	Warehouse
Dec 30 - Jan 7	<0.02	<0.03	<0.04	<0.03	<0.02	<0.02	<0.03	<0.02	0.04±0.02	<0.02	0.04±0.02
Jan 7 - Jan 14	<0.04	<0.04	<0.06	<0.04	<0.04	<0.05	<0.04	<0.05	<0.05	<0.04	<0.04
Jan 14 - Jan 22	<0.04	<0.04	<0.04	<0.04	<0.04	(a)	<0.04	<0.04	(a)	<0.04	<0.04
Jan 22 - Jan 29	<0.05	<0.05	<0.05	<0.05	<0.05	<0.02 (b)	<0.05	0.06±0.03	<0.02 (b)	<0.05	<0.05
Jan 29 - Feb 5	<0.03	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	0.04±0.02	0.04±0.02
Feb 5 - Feb 12	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	0.06±0.03	0.06±0.03	<0.04	0.16±0.03
Feb 12 - Feb 19	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	0.05±0.03	0.05±0.03	<0.04	0.07±0.04 (c)
Feb 19 - Feb 26	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04
Feb 26 - Mar 4	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04
Mar 4 - Mar 11	<0.04	<0.05	<0.17 (c)	<0.04	<0.05	<0.04	<0.05	<0.04	<0.05	<0.04	<0.04
Mar 11 - Mar 18	<0.05	<0.04	(d)	<0.04	<0.04	<0.04	<0.06 (c)	<0.05	<0.05	<0.04	<0.04
Mar 18 - Mar 25	<0.05	<0.04	<0.05	<0.04	<0.04	<0.04	0.05±0.03	<0.04	<0.05	<0.04	<0.04
Mar 25 - Apr 1	<0.04	0.06±0.03	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	0.04±0.03
Apr 1 - Apr 8	<0.05	<0.05	<0.05	<0.05	<0.05	<0.04	<0.05	<0.05	<0.06 (c)	<0.05	<0.04
Apr 8 - Apr 15	<0.04	<0.05	<0.04	<0.04	<0.05	(d)	<0.05	<0.04	<0.05	<0.04	<0.04
Apr 15 - Apr 22	<0.04	<0.05	<0.05	<0.05	<0.05	<0.04	<0.05	<0.05	<0.05	<0.05	<0.05
Apr 22 - Apr 29	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04
Apr 29 - May 6	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04
May 6 - May 13	<0.05	<0.05	<0.05	<0.05	<0.05	<0.05	<0.05	<0.05	<0.05	<0.05	<0.05
May 13 - May 20	0.05±0.03	<0.04	<0.04	<0.04	<0.04	<0.04	<0.04	<0.07 (c)	<0.04	<0.04	<0.04
May 20 - May 27	<0.04	<0.04	<0.04	<0.04	<0.05	<0.04	<0.05	<0.04	<0.04	<0.04	<0.04
May 27 - June 3	<0.05	<0.05	<0.05	<0.05	<0.05	<0.04	<0.05	<0.05	<0.05	<0.04	<0.05
June 3 - June 10	<0.05	<0.05	<0.05	<0.05	<0.05	<0.04	<0.05	<0.05	<0.05	<0.05	<0.05
June 10 - June 16	<0.06	<0.06	<0.06	<0.06	<0.06	(a)	<0.07 (c)	<0.06	<0.06	<0.18 (c)	<0.06
June 16 - June 24	<0.06	<0.06	<0.06	<0.06	<0.06	<0.03 (b)	<0.06	<0.06	<0.06	<0.06	<0.06
June 24 - July 1	<0.07	<0.07	<0.07	<0.07	<0.07	<0.06	<0.07	<0.07	<0.07	<0.07	<0.06

(a) Instrument station inaccessible
(b) Two week sample
(c) Not full week sample
(d) Instrument malfunction

TABLE A-2

SEMI-ANNUAL SUMMARY OF RADIOACTIVE GASEOUS EFFLUENTS
1976

	Quarter 1	Quarter 2
Gross Radioactivity		
Total Released (Ci)	1.14E 4	2.02E 4
Avg. Release Rate (uCi/sec)	1.45E 3	2.57E 3
Tritium		
Total Released (Ci)	9.33E 0	3.48E 0
Avg. Release Rate (uCi/sec)	1.19E 0	4.43E-1
Gross Alpha Radioactivity (Ci)	2.62E-6	1.33E-6
Isotopes Released (all Curies)		
Halogens-Main Stack		
Iodine-131	2.10E-2	3.61E-3
Iodine-133	3.29E-2	1.50E-2
Iodine-135	4.46E-2	2.16E-2
Halogens-Reactor Bldg. Vent		
Iodine-131	1.89E-1	6.09E-3
Iodine-133	5.86E-2	2.11E-2
Iodine-135	1.05E-1	3.69E-2
Particulates-Main Stack		
Strontium-89	4.74E-3	6.57E-3
Strontium-90	3.50E-5	3.84E-5
Cesium-134	1.10E-5	5.46E-6
Cesium-137	1.31E-4	9.89E-5
Barium-Lanthanum-140	4.79E-3	2.10E-2
Manganese-54	2.02E-5	9.70E-6
Cobalt-58	2.63E-6	2.44E-6
Cobalt-60	4.10E-5	3.66E-5
Zirconium-Niobium-95	5.10E-6	4.74E-6
Cerium-141	4.04E-6	2.29E-6
Cerium-144	2.13E-6	1.75E-6
Iron-59	4.22E-6	1.07E-8
Zinc-65	2.33E-6	NDA
Particulates-Reactor Bldg. Vent		
Strontium-89	1.38E-3	2.42E-3
Strontium-90	7.75E-6	4.65E-6
Cesium-134	2.31E-4	1.07E-4
Cesium-137	8.87E-4	3.20E-4

TABLE A-2 (Continued)

	Quarter 1	Quarter 2
Barium-Lanthanum-140	1.80E-3	5.46E-3
Magnanese-54	1.06E-3	2.39E-4
Cobalt-58	1.80E-4	4.91E-5
Cobalt-60	3.39E-3	7.93E-4
Zinc-65	1.13E-4	3.16E-5
Zirconium-Niobium-95	5.27E-4	7.33E-5
Cerium-141	1.38E-4	1.03E-5
Cerium-144	3.95E-4	6.74E-5
Iron-59	1.95E-4	8.84E-6
Silver-110m	1.62E-4	NDA
Chromium-51	NDA	7.59E-6
Gases[(a)]		
Main Stack	5.41E 3	1.39E 4
Reactor Bldg. Vent	5.99E 3	6.33E 3

(a) Unable to determine isotopes due to Augmented Off-Gas System

NDA = No detectable activity

BOSTON EDISON COMPANY

PILGRIM NUCLEAR GENERATING STATION

Environmental Radiation Monitoring Program

SEMIANNUAL REPORT NO. 9
JULY 1, 1976 THROUGH DECEMBER 31, 1976

Prepared By

Joel I. Cehn
Environmental Sciences Group
Nuclear Engineering Department

March 1977

Approved By: G. James Davis

G. James Davis, Group Leader
Environmental Sciences

I. INTRODUCTION AND SUMMARY

This report discusses the data accumulated by the Environmental Monitoring Program during the period July 1, through December 31, 1976.

Pilgrim Station reactor power output was about 75% of capacity during July, August and September. The average capacity factor for October was about 70%, and for November and December it was about 80%.

The Environmental Radiation Monitoring Program was performed during the reporting period as required by the Nuclear Regulatory Commission. The most notable event was the detection of weapons test fallout from the Chinese tests of September 26 and November 17. Isotopes detected in terrestrial media were I-131, Mn-54, Zr/Nb-95, Ba/La-140 and Ru-103, 106. I-131 was detected in milk at 51 pCi/l. Fallout nuclides detected in marine media included Mn-54, Zr/Nb-95, Ru-103, La-140 and Ce-141, 144.

Plant related radioactivity was detected in marine algae, sediments and finfish. The highest level detected was 2.1$\pm$ .2 pCi Co-60 per gram Irish Moss.

TABLE 6

AIR PARTICULATES - GROSS BETA CONCENTRATIONS IN WEEKLY SAMPLES (pCi/m^3)[(a)] July - Dec 1976

Period (1976)	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property Line	Overlook Area	Pedestrian Bridge	East Breakwater	Rocky Hill Road (East)	Warehouse
July 1 - July 8	0.022	0.027	0.014	0.028	0.030	0.026	0.027	0.029	0.027	0.028	0.022
July 8 - July 15	0.028	0.031	0.027	0.032	0.028	0.031	0.027	0.030	0.030	0.028	0.031
July 15 - July 22	0.025	0.034	0.023	0.024	0.031	0.026	0.022	0.031	0.024	0.020	0.028
July 22 - July 29	0.032	0.034	0.027	0.025	(d)	0.033	0.030	0.039	0.045	0.028	0.045
July 29 - Aug 5	0.031	0.033	0.026	0.028	0.020	0.027	0.027	0.024	0.028	0.027	0.034
Aug 5 - Aug 12	0.022	0.022	0.018	0.025	0.021	0.019	0.007(e)	0.021	0.018	0.030	0.027
Aug 12 - Aug 19	0.034	0.023	0.022	0.024	0.023	0.022	0.025	0.024	(d)	0.024	0.027
Aug 19 - Aug 26	0.037	0.036	0.038	0.034	0.039	0.039	0.032	0.035	0.030	0.043	0.038
Aug 26 - Sep 2	0.025	0.025	0.025	0.028	0.031	0.041	0.027	0.039	0.034	0.034	0.039
Sep 2 - Sep 9	0.026	0.026	0.034	0.010(e)	0.023	0.026	0.018	0.030	0.025	0.021	0.018
Sep 9 - Sep 16	0.036	0.037	0.043	0.034	0.026	0.034	0.038	0.039	0.040	0.041	0.035
Sep 16 - Sep 23	0.040	0.034	0.027	0.031	0.039	0.032	0.033	0.046	0.040	0.060	0.037
Sep 23 - Sep 30	0.039	0.044	0.031	0.037	0.037	0.033	0.030	0.038	(b)	0.028	0.043
Sep 30 - Oct 7	0.117	0.178	0.269	0.282	0.290	0.210	0.223	0.145	0.096(c)	0.168	0.181
Oct 7 - Oct 14	0.565	0.399	0.389	0.289	0.330	0.441	0.498	0.395	0.433	0.296	0.417
Oct 14 - Oct 21	0.100	0.088	0.122	0.120	0.130	0.142	0.092	0.133	2.590	0.103	0.135
Oct 21 - Oct 28	0.103	0.159	0.134	0.072	0.082	0.183	0.096	0.116	0.105	0.140	0.146
Oct 28 - Nov 4	0.202	0.346	0.260	0.278	0.391	(d)	0.262	0.302	0.231	0.250	0.297
Nov 4 - Nov 10	0.133	0.133	0.115	0.101	0.144	(d)	0.111	0.125	0.141	0.141	0.208
Nov 10 - Nov 18	0.195	0.180	0.135	0.143	0.178	0.084	0.163	0.161	0.162	0.140	0.134
Nov 18 - Nov 24	0.141	0.128	0.116	0.127	0.138	0.112	0.142	0.118	0.131	0.123	0.147
Nov 24 - Dec 2	0.145	0.137	0.094	0.140	0.12 (f)	0.100	0.13 (f)	0.129	0.08 (f)	0.14 (f)	0.10 (f)
Dec 2 - Dec 9	0.076	0.071	0.079	0.062	0.069	0.079	0.061	0.083	0.069	0.101	0.132
Dec 9 - Dec 16	0.096	0.132	0.094	0.090	0.101	0.102	0.098	0.093	0.103	0.099	0.105
Dec 16 - Dec 23	0.048	0.078	0.047	0.073	0.068	0.063	0.071	0.081	0.071	0.064	(d)
Dec 23 - Dec 30	(b)	0.062	0.023	0.069	0.072	0.059	(b)	0.062	(b)	0.07	0.077

(a) Results of weekly beta counting of air particulate filters. Measurement error (2σ) is typically 0.008 pCi/m^3, ranging from 0.006 to 0.014.
(b) Station inaccessible
(c) Two week sample
(d) Station inoperable
(e) Low sample volume - instrument malfunction
(f) Average of two air samples taken sequentially during period

TABLE 7

AIR PARTICULATES - GROSS GAMMA CONCENTRATION IN MONTHLY COMPOSITES [a]

(cpm/m^3 x 10^{-3})

Collection Period (1976)	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property Line	Overlook Area	Pedestrian Bridge	East Breakwater	Rocky Hill Road (East)	Warehouse
July	5±3	5±3	5±3	7±3	<4	6±3	7±3	9±3	7±3	5±3	8±3
August	5±2	<2	2±2	7±2	9±2	5±2	<2	2±2	<3	3±2	6±2
September	<3	<6	<3	<3	<3	<3	<6	6±3	<4	<3	<3
October	21±3	21±3	12±3	17±3	17±3	19±3	15±3	18±3	107±3	17±3	10±3
November	23±2	21±3	24±2	16±2	20±3	32±4	20±3	18±2	20±2	18±3	24±2
December	<4	4±3	16±3[a]	<3	17±3[a]	<3	<4	<3	20±4[a]	6±3	16±4[a]

(a) No isotopes identified from gamma isotopic spectrum.

TABLE 7A

AIR PARTICULATES - GAMMA ISOTOPE CONCENTRATION IN MONTHLY COMPOSITES[(a)]

(pCi/m^3)

Collection Period (1976)	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property Line	Overlook Area	Pedestrian Bridge	East Breakwater	Rocky Hill Road (East)	Warehouse
July											
Cs-137	0±.006	0±.007	0±.007	0±.007	0±.010	0±.007	0±.008	0±.006	0±.007	0±.007	0±.006
Mn-54	0±.006	0±.008	0±.008	0±.006	0±.009	0±.007	0±.006	0±.006	0±.007	0±.007	0±.007
Co-58	0±.006	0±.009	0±.008	0±.008	0±.011	0±.007	0±.008	0±.007	0±.007	0±.009	0±.007
Co-60	0±.007	0±.009	0±.008	0±.006	0±.011	0±.008	0±.008	0±.007	0±.007	0±.006	0±.007
Zn-65	0±.013	0±.02	0±.02	0±.015	0±.02	0±.02	0±.02	0±.011	0±.015	0±.02	0±.013
K-40	0±.09	0±.12	0±.120	0±.07	.1±.2	0±.11	0±.12	0±.06	0±.09	0±.10	0±.10
Ra-226	0±.011	0±.013	0±.012	0±.010	0±.02	0±.012	0±.012	0±.011	0±.012	0±.013	0±.011
Th-228	0±.008	.005±.009	0±.010	0±.008	0±.013	0±.009	0±.009	0±.008	0±.009	0±.010	.007±.008
Be-7	0±.06	.12±.09	.07±.070	.08±.05	.08±.06	.08±.07	.05±.07	0±.06	.09±.05	.11±.07	.10±.06
Analysis Date	8/6/76	8/10/76	8/4/76	8/18/76	8/5/76	8/6/76	8/9/76	8/9/76	8/5/76	8/9/76	8/5/76
October											
Cs-137	0±.007	0±.009	0±.008	0±.008	0±.008	0±.006	0±.009	0±.006	0±.006	0±.008	0±.008
Mn-54	.006±.006	0±.008	0±.007	0±.007	0±.008	0±.007	0±.008	0±.006	0±.007	0±.009	0±.008
Co-58	0±.010	0±.012	0±.010	0±.010	0±.010	0±.009	0±.012	0±.009	0±.009	0±.010	0±.011
Co-60	0±.007	0±.009	0±.008	0±.008	0±.008	0±.007	0±.010	0±.007	0±.007	0±.008	0±.010
Zn-65	0±.02	0±.02	0±.02	0±.02	0±.02	0±.02	0±.02	0±.02	0±.014	0±.02	0±.02
Zr-95	.02±.02	.03±.02	.02±.02	.02±.02	0±.02	.01±.02	.03±.02	0±.02	.03±.02	.02±.02	0±.02
Nb-95	.026±.012	.02±.02	.015±.011	.009±.015	0±.02	.022±.014	.020±.014	0±.02	.011±.010	.02±.02	0±.02
K-40	0±.08	0±.12	0±.11	0±.13	.02±.06	0±.09	0±.13	0±.10	0±.08	0±.11	.01±.05
Ra-226	0±.012	0±.015	0±.013	0±.014	0±.014	0±.011	0±.014	0±.012	0±.013	0±.013	0±.014
Th-228	0±.010	0±.010	0±.010	0±.010	0±.010	0±.009	0±.011	0±.009	0±.010	0±.010	0±.010
Be-7	.07±.07	.12±.10	.09±.08	.12±.06	.15±.08	0±.08	.07±.10	.10±.07	0±.08	.07±.09	.10±.07
Ba-140	0±.2	0±.2	.2±.2	.1±.2	0±.2	.22±.14	.1±.2	0±.2	0±.2	0±.2	0±.2
La-140	.007±.006	.006±.006	.008±.007	.008±.007	.008±.008	.006±.006	.010±.012	.007±.006	.006±.008	.008±.007	.009±.008
Other			.016±.011 Ru-103	.012±.014 Ru-103	.019±.014 Ru-103			.022±.010 Ru-103	.37±.08 Ru-106		
Analysis Date	11/22/76	11/22/76	11/19/76	11/22/76	11/19/76	11/22/76	11/23/76	11/24/76	11/19/76	11/24/76	11/18/76

(a) Results of Ge(Li) gamma spectrometry.

TABLE 7B

AIR PARTICULATES - STRONTIUM 89,90 CONCENTRATION IN QUARTERLY COMPOSITES(a) ($pCi/m^3 \times 10^{-4}$)

Collection Period (1976)	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property Line	Overlook Area	Pedestrian Bridge	East Breakwater	Rocky Hill Road (East)	Warehouse
Jan.-March											
Sr-89	<9	<25	<11	<18	<12	<7	15±12	12±10	<11	<9	<11
Sr-90	2±1	<3	<1	3±1	4±1	2±1	<1	2±1	2±1	2±1	<1
April-June											
Sr-89	<8	<9	<10	6±6	<12	7±7	<10	<9	<8	<9	<8.4
Sr-90	3.6±1.5	5±2	3±2	<11	2±2	2.7±1.4	4±2	4±2	5.9±1.5	2.7±1.5	1.9±1.3
July-Sept.											
Sr-89	<20	<6	9±6	11±11	<12	<14	<9	<30	<20	<9.5	<9.1
Sr-90	<2	2.0±1.2	1.9±0.8	2.4±1.2	<1.5	3±2	2.6±1.2	3±3	3±2	3.6±1.1	3.0±1.1
Oct.-Dec.											
Sr-89	<7	30±20		80±30	60±20	220±20	90±20	90±20	100±30	100±20	90±30
Sr-90	1.6±0.8	<2		<5	<3	<2	2±2	2±2	<4	2±2	<4

(a) Results of analyses per Regulatory Guide 4.6.

TABLE 8

PARTICULATE IODINE-131 IN AIR SAMPLES (pCi/m^3)[(a)]

July - December

Collection Period (1976)	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property Line	Overlook Area	Pedestrian Bridge	East Breakwater	Rocky Hill Road (East)	Warehouse
Jul 1-Jul 8	.00±.02	-.01±.02	.01±.02	.00±.02	.00±.02	-.01±.02	-.01±.02	.00±.02	.00±.02	.01±.02	.01±.02
Jul 8-Jul 15	.00±.02	.00±.02	.00±.02	.00±.02	.01±.02	.00±.02	.00±.02	.00±.02	.00±.02	.01±.02	.01±.02
Jul 15-Jul 22	.00±.04	-.02±.04	-.01±.04	-.02±.03	.00±.04	-.01±.04	-.02±.03	.00±.04	.00±.04	-.01±.04	-.01±.04
Jul 22-Jul 29	.00±.02	-.01±.02	.00±.02	.00±.02	(d)	.00±.02	-.01±.02	.00±.02	-.01±.02	.00±.02	-.01±.02
Jul 29-Aug 5	-.01±.02	-.01±.02	.00±.02	.00±.02	.00±.02	.00±.02	-.01±.02	-.01±.02	.00±.02	-.01±.02	.00±.02
Aug 5-Aug 12	.02±.04	.02±.04	.01±.04	-.01±.04	.01±.04	.03±.04	.05±.07(e)	.03±.04	.03±.04	.01±.04	.01±.04
Aug 12-Aug 19	.01±.02	.00±.02	.01±.02	-.01±.02	.00±.02	.015±.014	.00±.02	.02±.02	(d)	.01±.02	.01±.02
Aug 19-Aug 26	.02±.03	.01±.03	.01±.03	.01±.03	.00±.03	.00±.02	-.01±.03	.02±.03	.01±.03	.00±.02	.01±.03
Aug 26-Sep 2	.00±.02	.01±.02	.01±.02	.01±.02	.018±.015	.026±.014	.02±.02	.022±.014	.00±.02	.023±.014	.028±.015
Sep 2-Sep 9	.02±.04	.03±.04	.00±.04	.03±.05(e)	.05±.02	.04±.02	.04±.03	.04±.03	.04±.02	.00±.03	.03±.04
Sep 9-Sep 16	.02±.04	.04±.03	.02±.04	.01±.03	.02±.04	.04±.02	.02±.04	.06±.02	.00±.04	.00±.04	.01±.04
Sep 16-Sep 23	-.016±.014	.00±.02	.00±.02	.00±.02	-.015±.014	.00±.02	-.01±.02	.01±.02	-.01±.02	-.01±.02	-.01±.02
Sep 23-Sep 30	.01±.02	.01±.02	.017±.013	.01±.02	.01±.02	.01±.02	.01±.02	.018±.013	(b)	.01±.02	.01±.02
Sep 30-Oct 7	.01±.02	.03±.02	.06±.02	.05±.02	.05±.02	.05±.02	.06±.02	.04±.02	.018±.008(c)	.04±.02	.06±.02
Oct 7-Oct 14	.13±.02	.08±.02	.08 .02	.08±.02	.08±.02	.12±.02	.09±.02	.09±.02	.09±.02	.06±.02	.08±.02
Oct 14-Oct 21	.018±.012	.034±.013	.019±.013	.035±.012	.032±.012	.040±.012	.034±.013	.044±.012	0.0±.02(g)	.032±.012	.034±.013
Oct 21-Oct 28	.00±.02	.01±.02	.017±.014	.01±.02	.00±.02	.018±.013	.01±.02	.018±.014	.01±.02	.023±.015	.016±.014
Oct 28-Nov 4	.027±.015	.035±.015	.039±.015	.031±.014	.051±.015	(d)	.03±.02	.036±.015	.046±.015	.03±.02	.05±.02
Nov 4-Nov 10	.00±.05	.05±.03	.02±.05	.04±.05	.05±.03	(d)	.04±.05	.05±.03	.04±.05	.05±.03	.06±.03
Nov 10-Nov 18	.024±.013	.022±.013	.024±.013	.01±.02	.025±.014	.01±.02	.017±.013	.018±.013	.027±.013	.01±.02	.019±.013
Nov 18-Nov 24	.01±.03	.03±.04	.03±.02	.04±.02	.02±.04	-.01±.03	.00±.03	-.01±.03	.01±.03	.01±.04	.04±.02
Nov 24-Dec 2	.017±.013	.01±.02	.01±.02	.022±.012	.02±.08(f)	.01±.02	-.01±.08(f)	.024±.013	.03±.05(f)	.02±.08(f)	.02±.06(f)
Dec 2-Dec 9	-.01±.02	.00±.02	.01±.02	.00±.02	.01±.02	.02±.02	.00±.02	.01±.02	.00±.02	.00±.02	.02±.02
Dec 9-Dec 16	.00±.02	.00±.02	.00±.02	.02±.02	.01±.02	.01±.02	.01±.02	.01±.02	.00±.02	.01±.02	-.01±.02
Dec 16-Dec 23	.01±.02	.00±.02	.01±.02	.00±.02	.00±.02	-.01±.02	.01±.02	.01±.02	.01±.02	.01±.02	(d)
Dec 23-Dec 30	(b)	.01±.04	.00±.04	.02±.04	.02±.04	.02±.04	(b)	.04±.03	(b)	.01±.04	.04±.03

(a) Results of discriminated gamma counting (at 364 keV) of air particulate filters. Results are given in "raw" form to permit averaging. Errors are $\pm 2\sigma$.
(b) Station inaccessible.
(c) Two-week sample.
(d) Station inoperable.
(e) Low sample volume instrument malfunction.
(f) Average of two air samples taken sequentially during period.
(g) Calculated from Ge(Li) gamma spectrum.

TABLE 8A

GASEOUS IODINE-131 IN AIR SAMPLES (pCi/m^3)[(a)] JULY - DECEMBER

	Control	Offsite			Onsite						
Collection Period (1976)	East Weymouth	Plymouth Center	Cleft Rock Area	Manomet Substation	Rocky Hill Road (West)	Property Line	Overlook Area	Pedestrian Bridge	East Breakwater	Rocky Hill Road (East)	Warehouse
Jul 1-Jul 8	-.01±.04	.00±.04	.02±.04	-.01±.04	.02±.04	.00±.04	.00±.04	.02±.04	.00±.04	-.01±.04	.00±.04
Jul 8-Jul 15	.00±.04	.01±.04	.02±.04	-.01±.04	.00±.04	.02±.04	-.01±.04	.02±.04	.01±.04	-.01±.04	.00±.04
Jul 15-Jul 22	.03±.05	.02±.05	.03±.05	.03±.05	.03±.05	.03±.05	.01±.04	.04±.05	.04±.05	.01±.04	.01±.05
Jul 22-Jul 29	-.01±.04	-.02±.04	-.02±.04	.00±.04	(d)	-.01±.04	-.03±.04	-.05±.02	-.03±.05	.00±.04	.00±.04
Jul 29-Aug 5	-.03±.05	-.03±.05	.02±.05	-.02±.04	-.02±.05	.01±.05	-.02±.05	.03±.05	-.03±.05	-.02±.05	-.02±.05
Aug 5-Aug 12	.03±.09	.02±.08	.12±.06	-.01±.07	.02±.08	.08±.05	.01±.13(e)	.07±.08	.08±.06	.07±.08	.08±.05
Aug 12-Aug 19	.04±.03	.04±.05	.03±.04	.02±.04	.01±.04	.05±.03	.05±.03	.04±.03	(d)	.04±.03	.04±.03
Aug 19-Aug 26	.03±.05	-.03±.05	.01±.05	-.02±.05	-.03±.05	-.01±.05	.00±.05	.00±.05	-.03±.05	.04±.05	.01±.05
Aug 26-Sep 2	.01±.04	.02±.04	.03±.04	-.01±.04	.03±.04	.02±.04	.02±.05	.03±.04	.05±.03	.02±.04	.02±.04
Sep 2-Sep 9	.02±.04	.00±.05	.03±.05	.02±.06(e)	.00±.04	.03±.04	.02±.05	.02±.05	.01±.04	.03±.04	.03±.04
Sep 9-Sep 16	.05±.03	.03±.04	.05±.03	.03±.04	.04±.03	.03±.04	.03±.04	.03±.04	.04±.03	.05±.03	.04±.03
Sep 16-Sep 23	-.01±.04	.02±.04	.02±.04	.00±.04	.01±.04	.00±.04	.01±.04	-.02±.04	.00±.04	.02±.04	.02±.04
Sep 23-Sep 30	.02±.04	.01±.04	.03±.04	.03±.04	.01±.04	.04±.03	.03±.04	.05±.03	(b)	.02±.04	.03±.04
Sep 30-Oct 7	.10±.03	.02±.05	.57±.05	.10±.03	.26±.04	.09±.03	.10±.04	.08±.03	.02±.02(c)	.12±.03	.06±.03
Oct 7-Oct 14	.04±.03	.08±.03	.13±.03	.11±.03	.07±.03	.11±.03	.10±.03	.05±.03	.12±.03	.08±.03	.06±.03
Oct 14-Oct 21	.03±.04	.07±.03	.12±.03	.07±.02	.04±.02	.07±.02	.03±.04	.06±.02	.09±.02(g)	.08±.02	.04±.02
Oct 21-Oct 28	.02±.04	.03±.04	.06±.03	.02±.04	.00±.04	.01±.04	.00±.04	.04±.03	.04±.05	.03±.04	.04±.03
Oct 28-Nov 4	.05±.03	.02±.04	.06±.03	.04±.03	.02±.04	(d)	.02±.04	.04±.03	.04±.03	.05±.03	.05±.03
Nov 4-Nov 10	.07±.10	.05±.10	.07±.10	.07±.10	.06±.10	(d)	.04±.10	.05±.09	.09±.10	.10±.07	.06±.09
Nov 10-Nov 18	.01±.04	-.01±.04	.02±.04	.00±.04	.02±.04	.03±.05	-.01±.04	.01±.04	.02±.04	-.01±.04	-.01±.04
Nov 18-Nov 24	.00±.07	.01±.07	.00±.07	.01±.07	.03±.07	.03±.07	.05±.07	.04±.07	-.06±.07	-.02±.07	.03±.07(f)
Nov 24-Dec 2	.02±.04	.02±.04	.01±.04	.03±.04	.05±.13(f)	.01±.04	.07±.12(f)	.00±.04	.03±.13(f)	.07±.13(f)	.07±.12
Dec 2-Dec 9	-.01±.04	.02±.05	.00±.05	.03±.05	.01±.04	.00±.04	.02±.05	.02±.05	.02±.03	.02±.05	.09±.03
Dec 9-Dec 16	.01±.05	-.03±.04	-.01±.05	-.03±.04	.01±.05	-.03±.04	-.03±.05	.01±.05	.04±.05	.02±.04	-.02±.04
Dec 16-Dec 23	.04±.05	.01±.05	.02±.04	.02±.05	.02±.05	.00±.05	.02±.05	.00±.04	.04±.05	-.02±.04	(d)
Dec 23-Dec 30	(b)	.04±.09	.08±.09	.06±.09	.02±.09	.06±.09	(b)	.04±.09	(b)	.05±.09	.01±.08

(a) Results of discriminated gamma counting (at 364 keV) of gaseous iodine filters. Results are given in "raw" form to permit averaging. Errors are ±2σ.
(b) Station inaccessible.
(c) Two-week sample.
(d) Station inoperable.
(e) Low sample volume instrument malfunction.
(f) Average of two air samples taken sequentially during period.

TABLE IV-1

Gross Beta Activity in Air Particulates

Observed Concentrations vs. Plant Predicted Contributions[(a)]

(pCi/m^3)

Air Particulate Sampling Station	3rd Quarter[(b)] I[(d)] Observed	II[(e)] Predicted	II:I	4th Quarter[(c)] I[(d)] Observed	II[(e)] Predicted	II:I
East Weymouth (Background)	3.1×10^{-2}	2.4×10^{-5}	7.7×10^{-4}	1.6×10^{-1}	2.0×10^{-5}	1.2×10^{-4}
Plymouth Center (Offsite)	3.1×10^{-2}	2.2×10^{-4}	7.1×10^{-3}	1.6×10^{-1}	1.4×10^{-4}	8.8×10^{-4}
Cleft Rock Area (Offsite)	2.7×10^{-2}	4.5×10^{-3}	0.17	1.4×10^{-1}	3.5×10^{-3}	0.02
Manomet Substation (Offsite)	3.0×10^{-2}	5.9×10^{-4}	0.02	1.4×10^{-1}	9.7×10^{-3}	0.07
Rocky Hill Rd.(West) (Onsite)	2.9×10^{-2}	1.2×10^{-3}	0.04	1.6×10^{-1}	8.0×10^{-4}	5.0×10^{-3}
Property Line (Onsite)	3.0×10^{-2}	5.7×10^{-4}	0.02	1.4×10^{-1}	9.1×10^{-4}	6.5×10^{-3}
Overlook Area (Onsite)	2.8×10^{-2}	4.0×10^{-4}	0.01	1.6×10^{-1}	3.2×10^{-3}	0.02
Pedestrian Bridge (Onsite)	3.3×10^{-2}	1.7×10^{-3}	0.05	1.5×10^{-1}	1.5×10^{-2}	0.10
East Breakwater (Onsite)	3.1×10^{-2}	1.5×10^{-3}	0.05	3.5×10^{-1}(f)	1.3×10^{-2}	0.04
Rocky Hill Rd.(East) (Onsite)	3.2×10^{-2}	4.3×10^{-4}	0.01	1.4×10^{-1}	1.5×10^{-2}	0.11
Warehouse (Onsite)	3.3×10^{-2}	8.2×10^{-4}	0.02	1.7×10^{-1}	6.3×10^{-3}	0.04

(a) observed gross beta concentrations in air are average quarterly values for each air particulate monitoring station (see Table 6). The gross beta concentrations in air predicted from plant effluent data and site specific meteorology are included for comparison.

(b) 3rd quarter extends from July 1 to September 30, 1976.

(c) 4th quarter extends from September 30 to December 30, 1976.

(d) column I indicates average quarterly gross beta concentrations observed at each air sampling station (pCi/m^3)

(e) column II indicates predicted gross beta concentration of air particulate due to plant effluents

TABLE IV-2
Iodine-131 Air Sample Comparisons[(a)] July - December
(pCi/m^3)

Collection Period	CONTROL: East Weymouth I[(b)]	CONTROL: East Weymouth II[(c)]	OFF SITE: Plymouth Center I[(b)]	OFF SITE: Plymouth Center II[(c)]	OFF SITE: Cleft Rock Area I[(b)]	OFF SITE: Cleft Rock Area II[(c)]	OFF SITE: Manomet Substation I[(b)]	OFF SITE: Manomet Substation II[(c)]	ON SITE: Rocky Hill Rd(West) I[(b)]	ON SITE: Rocky Hill Rd(West) II[(c)]	ON SITE: Property Line I[(b)]	ON SITE: Property Line II[(c)]
Jul 1 - Jul 8	-.01±.04		-.01±.04		.03±.04		-.01±.04		.02±.04		-.01±.04	
Jul 8 - Jul 15	.00±.04		.01±.04		.02±.04		-.01±.04		.01±.04		.02±.04	
Jul 15 - Jul 22	.03±.06		.00±.06		.02±.06		.01±.06		.03±.06		.02±.06	
Jul 22 - Jul 29	-.01±.04		-.03±.04		-.02±.04		.00±.04		(b)		-.01±.04	
Jul 29 - Aug 5	-.04±.05		-.04±.05		.02±.05		-.02±.04		-.02±.05		.01±.05	
Aug 5 - Aug 12	.05±.10		.04±.08		.13±.07		-.02±.08		.03±.09		.11±.06	
Aug 12 - Aug 19	.05±.04	$1.4x10^{-5}$	.04±.05	$1.5x10^{-4}$	.04±.04	$3.6x10^{-3}$	.01±.04	$4.5x10^{-4}$	.01±.04	$1.1x10^{-3}$	.07±.03	$8.2x10^{-4}$
Aug 19 - Aug 26	.05±.06		-.02±.06		.02±.06		-.01±.06		-.03±.05		-.01±.05	
Aug 26 - Sep 2	.01±.04		.03±.04		.04±.04		.00±.04		.05±.04		.05±.04	
Sep 2 - Sep 9	.04±.06		.03±.06		.03±.06		.05±.08		.05±.04		.07±.04	
Sep 9 - Sep 16	.07±.05		.07±.05		.07±.05		.04±.05		.06±.04		.07±.04	
Sep 16 - Sep 23	-.03±.04		.02±.04		.02±.04		.00±.04		-.01±.04		.00±.04	
Sep 23 - Sep 30	.03±.04		.02±.04		.05±.04		.04±.04		.02±.04		.05±.04	
Sep 30 - Oct 7	.11±.04		.05±.05		.63±.05		.15±.04		.31±.04		.14±.04	
Oct 7 - Oct 14	.17±.04		.16±.04		.21±.04		.19±.04		.15±.04		.23±.04	
Oct 14 - Oct 21	.05±.04		.10±.03		.14±.03		.11±.02		.07±.02		.11±.02	
Oct 21 - Oct 28	.02±.04		.04±.04		.08±.03		.03±.04		.00±.04		.03±.04	
Oct 28 - Nov 4	.08±.03		.06±.04		.10±.04		.07±.03		.07±.04		(e)	
Nov 4 - Nov 10	.07±.11		.10±.10		.09±.11		.11±.11		.11±.10		(e)	
Nov 10 - Nov 18	.03±.04	$7.2x10^{-6}$	.01±.04	$5.2x10^{-5}$	.04±.04	$1.3x10^{-3}$	.01±.04	$3.6x10^{-3}$	.05±.04	$3.4x10^{-4}$	.04±.05	$3.9x10^{-4}$
Nov 18 - Nov 24	.01±.08		.04±.08		.03±.07		.05±.08		.05±.08		.02±.08	
Nov 24 - Dec 2	.04±.04		.03±.04		.02±.04		.05±.04		.07±.15		.02±.04	
Dec 2 - Dec 9	-.02±.04		.02±.05		.01±.05		.03±.05		.02±.04		.02±.04	
Dec 9 - Dec 16	.01±.05		-.03±.04		-.01±.05		-.01±.04		.02±.05		-.02±.04	
Dec 16 - Dec 23	.05±.05		.01±.05		.03±.04		.02±.05		.02±.05		-.01±.05	
Dec 23 - Dec 30	(d)		.05±.10		.08±.10		.08±.10		.04±.10		.08±.10	

(a) measured I-131 in air samples are total of particulate I-131 (Table 8) and gaseous I-131 (Table 8A) reported with 2 sigma errors.
(b) column I indicates observed air concentration of I-131 in pCi/m^3
(c) column II indicates quarterly average calculated air concentrations of I-131 (pCi/m^3) based on plant effluent data and site specific meteorology
(d) station inaccessible
(e) station inoperable

Arschutzel copy
plse return.

PILGRIM NUCLEAR GENERATING STATION

Radioactive Effluent and Waste Disposal Report
including
Radiological Impact on Humans

JANUARY 1 THROUGH JUNE 30, 1977

BY: NUCLEAR ENGINEERING DEPT.
ENVIRONMENTAL SCIENCES GROUP

DATE
SEPT. 1, 1977

BOSTON EDISON COMPANY

TABLE 1A

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT (   )

January through June, 1977

GASEOUS EFFLUENTS–SUMMATION OF ALL RELEASES

	Unit	Quarter 1	Quarter 2	Est. Total Error, %
A. Fission & activation gases				
1. Total release	Ci	1.57 E+5	1.82 E+5	3.00 E+1
2. Average release rate for period	μCi/sec	2.02 E+4	2.31 E+4	
3. Percent of Technical specification limit	%	8.37 E+0	9.52 E+0	
B. Iodines				
1. Total iodine-131	Ci	1.33 E-1	2.03 E-1	5.00 E+1
2. Average release rate for period	μCi/sec	1.71 E-2	2.58 E-2	
3 Percent of technical specification limit	%	2.41 E+0	3.47 E+0	
C. Particulates				
1. Particulates with half-lives >8 days	Ci	8.35 E-2	7.78 E-2	4.66 E+1
2. Average release rate for period	μCi/sec	1.07 E-2	9.90 E-3	
3. Percent of technical specification limit	%	1.25 E+0	5.93 E-1	
4. Gross alpha radioactivity	Ci	2.66 E-6	6.05 E-7	
D. Tritium				
1. Total release	Ci	2.54 E+1	2.12 E+1	6.42 E+1
2. Average release rate for period	μCi/sec	3.27 E+0	2.70 E+0	
3. Percent of technical specification limit	%	N/A E	N/A E	

TABLE 1B

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT
January through June, 1977
GASEOUS EFFLUENTS–ELEVATED RELEASE

Nuclides Released	Unit	CONTINUOUS MODE Quarter	CONTINUOUS MODE Quarter	BATCH MODE Quarter	BATCH MODE Quarter
1. Fission gases					
krypton-85	Ci	3.58 E+3	4.88 E+3	. E	. E
krypton-85m	Ci	1.83 E+4	1.87 E+4	. E	. E
krypton-87	Ci	1.32 E+4	9.68 E+3	. E	. E
krypton-88	Ci	3.60 E+4	3.12 E+4	. E	. E
xenon-133	Ci	5.38 E+4	6.12 E+4	. E	. E
xenon-135	Ci	1.49 E+4	3.25 E+3	. E	. E
xenon-135m	Ci	3.77 E+2	7.57 E+2	. E	. E
xenon-138	Ci	2.03 E+3	2.10 E+3	. E	. E
Others (specify)	Ci	. E	. E	. E	. E
xenon-133M	Ci	9.35 E+2	9.96 E+2	. E	. E
	Ci	. E	. E	. E	. E
unidentified	Ci	9.69 E+3	4.49 E+4	. E	. E
Total for period	Ci	1.53 E+5	1.78 E+5	. E	. E
2. Iodines					
iodine-131	Ci	5.83 E-2	9.60 E-2	. E	. E
iodine-133	Ci	1.08 E-1	2.56 E-1	. E	. E
iodine-135	Ci	2.37 E-1	2.84 E-1	. E	. E
Total for period	Ci	4.03 E-1	6.36 E-1	. E	. E
3. Particulates					
strontium-89	Ci	9.06 E-3	3.00 E-2	. E	. E
strontium-90	Ci	9.72 E-5	1.02 E-4	. E	. E
cesium-134	Ci	8.15 E-6	1.01 E-5	. E	. E
cesium-137	Ci	1.06 E-3	6.10 E-4	. E	. E
barium-lanthanum-140	Ci	3.57 E-2	3.32 E-2	. E	. E
Others (specify)	Ci	. E	. E	. E	. E
manganese - 54	Ci	2.59 E-5	9.04 E-6	. E	. E
cobalt - 60	Ci	1.63 E-4	1.70 E-4	. E	. E
unidentified	Ci	. E	. E	. E	. E
cerium - 141		3.00E-5	2.15 E-5		
cerium - 144		5.09E-4	1.65 E-4		
zirconium-NIOBIUM-95		1.07E-6	N.D.A.		

TABLE 1C

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT

January through June, 1977

GASEOUS EFFLUENTS–GROUND-LEVEL RELEASES

		CONTINUOUS MODE		BATCH MODE	
Nuclides Released	Unit	Quarter	Quarter	Quarter	Quarter

1. Fission gases

krypton-85	Ci	. E	. E	. E	. E
krypton-85m	Ci	4.73E+0	NDA E	. E	. E
krypton-87	Ci	3.29E+1	NDA E	. E	. E
krypton-88	Ci	1.10E+1	NDA E	. E	. E
xenon-133	Ci	1.09E+3	1.79E+2	. E	. E
xenon-135	Ci	2.13E+2	5.19E+2	. E	. E
xenon-135m	Ci	1.29E+3	2.95E+3	. E	. E
xenon-138	Ci	1.22E+3	NDA E	. E	. E
Others (specify)	Ci	. E	. E	. E	. E
	Ci	. E	. E	. E	. E
	Ci	. E	. E	. E	. E
unidentified	Ci	. E	. E	. E	. E
Total for period	Ci	3.86 E+3	3.65E+3	. E	. E

2. Iodines

iodine-131	Ci	7.44 E-2	1.07E-1	. E	. E
iodine-133	Ci	2.64 E-1	2.18E-1	. E	. E
iodine-135	Ci	5.41 E-1	5.52E-1	. E	. E
Total for period	Ci	8.79 E-1	8.77E-1	. E	. E

3. Particulates

strontium-89	Ci	3.86 E-3	5.46 E-3	. E	. E
strontium-90	Ci	1.64E-5	1.52 E-5	. E	. E
cesium-134	Ci	3.86E-4	5.46 E-4	. E	. E
cesium-137	Ci	1.08 E-3	9.01 E-4	. E	. E
barium-lanthanum-140	Ci	2.94 E-2	5.60 E-3	. E	. E
Others (specify)	Ci	. E	. E	. E	. E
manganese - 54	Ci	4.96E-4	1.42E-4	. E	. E
cobalt - 60	Ci	6.63E-4	6.86E-4	. E	. E
unidentified	Ci	. E	. E	. E	. E
cerium - 141		7.46E-4	2.53E-5		
cerium - 144		2.28E-4	NDA		
chromium - 51		NDA	7.60E-5		

PILGRIM NUCLEAR POWER STATION

RADIOACTIVE EFFLUENT AND WASTE DISPOSAL REPORT

INCLUDING RADIOLOGICAL IMPACT ON HUMANS

JULY 1 THROUGH DECEMBER 31, 1977

Prepared by: Thomas L Sowdon

Thomas L. Sowdon
Radiological Engineer

Reviewed by: Susan L. Kannenberg

Susan L. Kannenberg
Senior Environmental Scientist

Approved by: Fred J. Mogolesko

Fred J. Mogolesko
Manager of Environmental Science Group

Date of Submittal: March 1, 1978

TABLE 1A

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT (1977)

GASEOUS EFFLUENTS - SUMMATION OF ALL RELEASES

July – December, 1977

	Unit	Quarter 3	Quarter 4	Est. Total Error, %
A. Fission and activation gases				
1. Total release	Ci	6.87E+4	5.30E+3	2.50E0
2. Average release rate for period	μCi/sec	8.64E+3	6.67E+2	
3. Percent of Technical Specification limit	%	3.51E0	3.75E−1	
B. Iodines				
1. Total iodine-131	Ci	1.44E−1	1.50E−3	2.81E+1
2. Average release rate for period	μCi/sec	1.81E−2	1.89E−4	
3. Percent of Technical Specification limit	%	2.28E0	1.13E−2	
C. Particulates				
1. Particulates with half-lives > 8 days	Ci	3.87E−2	8.25E−3	2.84E+1
2. Average release rate for period	μCi/sec	4.87E−3	1.04E−3	
3. Percent of Technical Specification limit	%	2.93E−1	1.07E−1	
4. Gross alpha radioactivity	Ci	3.89E−7	< 1.14E−7	
D. Tritium				
1. Total release	Ci	1.04E+1	3.71E0	3.40E+1
2. Average release rate for period	μCi/sec	1.31E0	4.67E−1	
3. Percent of Technical Specification limit	%	—	—	

TABLE 1B
EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT (YEAR)
GASEOUS EFFLUENTS – ELEVATED RELEASE

July – Dec. mber, 1977

		CONTINUOUS MODE		BATCH MODE	
Nuclides Released	Unit	Quarter	Quarter	Quarter	Quarter

1. Fission gases

krypton-85	Ci	2.36E0	3.07E−3		
krypton-85m	Ci	8.22E+3	6.61E+2		
krypton-87	Ci	6.12E+3	5.56E+2		
krypton-88	Ci	1.61E+4	1.46E+3		
xenon-133	Ci	2.90E+4	2.13E+2		
xenon-135	Ci	1.22E+3	5.64E+1		
xenon-135m	Ci	2.72E+2	8.46E+1		
xenon-138	Ci	9.51E+2	2.82E+2		
xenon-131m	Ci	1.77E+3	7.21E+2		
xenon-137	Ci	3.81E+3	NDA		
xenon-133m	Ci	4.75E+2	NDA		
Total for period	Ci	6.79E+4	4.03E+3		

2. Iodines

iodine-131	Ci	7.45E−2	1.24E−3		
iodine-133	Ci	6.96E−2	1.06E−2		
iodine-135	Ci	9.80E−2	1.75E−2		
Total for period	Ci	2.24E−1	2.93E−2		

3. Particulates

strontium-89	Ci	9.84E−3	3.06E−3		
strontium-90	Ci	5.60E−5	2.25E−5		
cesium-134	Ci	5.42E−5	2.86E−5		
cesium-137	Ci	2.65E−4	1.08E−4		
barium-lanthanum-140	Ci	2.13E−2	1.40E−3		
chromium-51	Ci	2.61E−5	NDA		
manganese-54	Ci	4.27E−5	1.03E−4		
cobalt-58	Ci	1.78E−6	2.33E−6		
iron-59	Ci	NDA	6.79E−6		
cobalt-60	Ci	1.76E−4	3.57E−4		
zinc-65	Ci	NDA	7.44E−6		
zirconium-niobium-95	Ci	NDA	1.79E−5		
cerium-141	Ci	2.08E−5	7.82E−6		
cerium-144	Ci	NDA	4.55E−6		
ruthenium-103	Ci	2.00E−6	1.76E−6		
ruthenium-106	Ci	1.50E−5	NDA		

TABLE 1C

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT (1977)

GASEOUS EFFLUENTS - GROUND LEVEL RELEASE

July – December, 1977

Nuclides Released	Unit	CONTINUOUS MODE Quarter	CONTINUOUS MODE Quarter	BATCH MODE Quarter	BATCH MODE Quarter
1. Fission gases					
krypton-85	Ci	NDA	NDA		
krypton-85m	Ci	NDA	NDA		
krypton-87	Ci	NDA	NDA		
krypton-88	Ci	NDA	NDA		
xenon-133	Ci	5.49E+2	9.18E+2		
xenon-135	Ci	7.00E+1	1.17E+2		
xenon-135m	Ci	4.18E+1	6.99E+1		
xenon-138	Ci	9.96E+1	1.67E+2		
Total for period	Ci	7.60E+2	1.27E+3		
2. Iodines					
iodine-131	Ci	6.99E–2	2.60E–4		
iodine-133	Ci	9.41E–2	1.06E–3		
iodine-135	Ci	1.44E–1	1.39E–3		
Total for period	Ci	3.08E–1	2.71E–3		
3. Particulates					
strontium-89	Ci	2.98E–3	2.98E–4		
strontium-90	Ci	8.13E–6	5.67E–6		
cesium-134	Ci	1.06E–4	9.60E–5		
cesium-137	Ci	2.76E–4	2.12E–4		
barium-lanthanum-140	Ci	2.18E–3	3.95E–4		
manganese-54	Ci	1.84E–4	4.33E–4		
cobalt-58	Ci	2.48E–5	1.52E–5		
iron-59	Ci	1.89E–5	NDA		
cobalt-60	Ci	9.27E–4	1.57E–3		
zinc-65	Ci	1.55E–5	1.06E–5		
zirconium-niobium-95	Ci	8.38E–6	9.47E–6		
cerium-141	Ci	3.04E–5	NDA		
ruthenium-103	Ci	5.83E–6	4.97E–6		
ruthenium-106	Ci	NDA	6.88E–5		

PILGRIM NUCLEAR POWER STATION

RADIOACTIVE EFFLUENT AND WASTE DISPOSAL REPORT

INCLUDING RADIOLOGICAL IMPACT ON HUMANS

JANUARY 1 THROUGH JUNE 30, 1978

Prepared by: ______________________
Thomas L. Sowdon
Senior Radiological Engineer

Approved by: ______________________
Fred J. Mogolesko
Manager of Environmental Science Group

Date of Submittal: September 1, 1978

TABLE 1A
EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT (1978)
GASEOUS EFFLUENTS - SUMMATION OF ALL RELEASES
January – June, 1978

	Unit	Quarter 1	Quarter 2	Est. Total Error, %
A. Fission and activation gases				
1. Total release	Ci	1.67E+4	8.92E+3	2.75E+1
2. Average release rate for period	μCi/sec	2.15E+3	1.13E+3	
3. Percent of Technical Specification limit	%	8.96E−1	4.98E−1	
B. Iodines				
1. Total iodine-131	Ci	1.96E−2	3.80E−2	3.91E+1
2. Average release rate for period	μCi/sec	2.57E−3	4.83E−3	
3. Percent of Technical Specification limit	%	3.23E−1	3.92E−1	
C. Particulates				
1. Particulates with half-lives > 8 days	Ci	1.77E−2	1.77E−2	3.75E+1
2. Average release rate for period	μCi/sec	2.28E−3	2.25E−3	
3. Percent of Technical Specification limit	%	1.99E−1	1.81E−1	
4. Gross alpha radioactivity	Ci	< 7.10E−7	< 2.72E−6	
D. Tritium				
1. Total release	Ci	1.12E+1	2.06E+1	4.90E+1
2. Average release rate for period	μCi/sec	1.44E0	2.62E0	
3. Percent of Technical Specification limit	%	——	——	

TABLE 1B

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT (1978)

GASEOUS EFFLUENTS – ELEVATED RELEASE

January - June, 1978

		CONTINUOUS MODE		BATCH MODE	
Nuclides Released	Unit	Quarter	Quarter	Quarter	Quarter

1. Fission gases

krypton-85	Ci	4.98E−2	3.22E−2	——	——
krypton-85m	Ci	2.83E+3	1.83E+3	——	——
krypton-87	Ci	2.66E+3	8.43E+2	——	——
krypton-88	Ci	6.77E+3	3.37E+3	——	——
xenon-133	Ci	1.94E+3	1.27E+3	——	——
xenon-135	Ci	1.18E+3	1.21E+2	——	——
xenon-135m	Ci	2.24E+2	1.79E+2	——	——
xenon-138	Ci	6.85E+2	7.31E+2	——	——
xenon-131m	Ci				
xenon-137	Ci				
xenon-133m	Ci				
Total for period	Ci	1.63E+4	8.34E+3	——	——

2. Iodines

iodine-131	Ci	9.91E−3	2.76E−2	——	——
iodine-133	Ci	4.83E−2	1.49E−1	——	——
iodine-135	Ci	7.20E−2	1.93E−1	——	——
Total for period	Ci	1.30E−1	3.70E−1	——	——

3. Particulates

strontium-89	Ci	3.69E−3	4.48E−3	——	——
strontium-90	Ci	2.73E−5	2.83E−5	——	——
cesium-134	Ci	6.63E−6	7.34E−7	——	——
cesium-137	Ci	1.16E−4	1.42E−4	——	——
barium-lanthanum-140	Ci	7.66E−3	7.52E−3	——	——
manganese-54	Ci	1.67E−4	1.31E−4	——	——
cobalt-58	Ci	9.33E−7	——	——	——
cobalt-60	Ci	5.84E−4	2.92E−4	——	——
niobium-95	Ci	3.39E−6	4.41E−6	——	——
cerium-141	Ci	9.91E−7	1.07E−5	——	——
chromium-51	Ci	——	3.64E−5	——	——
cerium-144	Ci	——	7.73E−5	——	——
ruthenium-106	Ci	——	1.29E−4	——	——
	Ci				
	Ci				
	Ci				

TABLE 1C

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT (1978)

GASEOUS EFFLUENTS - GROUND LEVEL RELEASE

January – June, 1978

Nuclides Released	Unit	CONTINUOUS MODE Quarter	CONTINUOUS MODE Quarter	BATCH MODE Quarter	BATCH MODE Quarter
1. Fission gases					
krypton-85	Ci				
krypton-85m	Ci				
krypton-87	Ci				
krypton-88	Ci				
xenon-133	Ci				
xenon-135	Ci	4.45E+2	5.82E+2	—	—
xenon-135m	Ci				
xenon-138	Ci				
Total for period	Ci	4.45E+2	5.82E+2	—	—
2. Iodines					
iodine-131	Ci	9.73E−3	1.04E−2	—	—
iodine-133	Ci	7.21E−2	7.05E−2	—	—
iodine-135	Ci	1.37E−1	5.87E−2	—	—
Total for period	Ci	2.19E−1	1.41E−1	—	—
3. Particulates					
strontium-89	Ci	9.28E−4	4.78E−4	—	—
strontium-90	Ci	7.05E−6	3.28E−6	—	—
cesium-134	Ci	7.03E−5	6.43E−5	—	—
cesium-137	Ci	3.52E−4	1.88E−4	—	—
barium-lanthanum-140	Ci	1.69E−3	1.99E−3	—	—
chromium-51	Ci	4.49E−5	1.97E−4	—	—
manganese-54	Ci	4.25E−4	2.91E−4	—	—
cobalt-58	Ci	6.53E−6	3.34E−5	—	—
cobalt-60	Ci	1.81E−3	1.40E−3	—	—
zinc-65	Ci	1.12E−5	9.33E−6	—	—
zirconium-niobium-95	Ci	6.04E−6	1.94E−5	—	—
cerium-141	Ci	2.64E−5	3.05E−5	—	—
cerium-144	Ci	1.15E−5	7.54E−5	—	—
	Ci				

PILGRIM NUCLEAR POWER STATION

RADIOACTIVE EFFLUENT AND WASTE DISPOSAL REPORT

INCLUDING RADIOLOGICAL IMPACT ON HUMANS

JULY 1 THROUGH DECEMBER 31, 1978

Prepared by: Thomas L. Sowdon

Thomas L. Sowdon
Senior Radiological Engineer

Approved by: Fred J. Mogolesko

Fred J. Mogolesko
Group Leader of Environmental
Sciences Group

Date of Submittal: March 1, 1979

TABLE 1A

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT

GASEOUS EFFLUENTS - SUMMATION OF ALL RELEASES

JULY - DECEMBER 1978

	Unit	Quarter 3	Quarter 4	Est. Total Error, %
A. Fission and activation gases				
1. Total release	Ci	4.24 E+3	2.83 E+3	5.00 E+1
2. Average release rate for period	μCi/sec	5.33 E+2	3.56 E+2	
3. Percent of Technical Specification limit	%	2.33 E-1	1.54 E-1	
B. Iodines				
1. Total iodine-131	Ci	2.66 E-2	4.00 E-2	3.75 E+1
2. Average release rate for period	μCi/sec	3.35 E-3	5.03 E-3	
3. Percent of Technical Specification limit	%	1.33 E0	2.00 E0	
C. Particulates				
1. Particulates with half-lives > 8 days	Ci	9.84 E-3	1.11 E-2	3.75 E+1
2. Average release rate for period	μCi/sec	1.24 E-3	1.40 E-3	
3. Percent of Technical Specification limit	%	1.19 E-1	1.71 E-1	
4. Gross alpha radioactivity	Ci	<4.43 E-7	<5.10 E-7	
D. Tritium				
1. Total release	Ci	2.61 E+1	3.76 E+1	5.00 E+1
2. Average release rate for period	μCi/sec	3.28 E0	4.73 E0	
3. Percent of Technical Specification limit	%			

TABLE 1B
EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT (1978)
GASEOUS EFFLUENTS – ELEVATED RELEASE

JULY - DECEMBER, 1978

		CONTINUOUS MODE		BATCH MODE	
Nuclides Released	Unit	Quarter	Quarter	Quarter	Quarter

1. Fission gases

krypton-85	Ci	1.25 E-2	7.98 E-3		
krypton-85m	Ci	9.52 E+2	6.06 E+2		
krypton-87	Ci	5.06 E+2	2.27 E+2		
krypton-88	Ci	1.74 E+3	1.06 E+3		
xenon-133	Ci	5.86 E+2	4.07 E+2		
xenon-135	Ci	7.45 E+1	4.76 E+1		
xenon-135m	Ci	4.33 E+1	7.54 E+1		
xenon-138	Ci	8.02 E+1	2.61 E+2		
	Ci				
	Ci				
	Ci				
Total for period	Ci	3.98 E+3	2.68 E+3		

2. Iodines

iodine-131	Ci	1.69 E-2	2.44 E-2		
iodine-133	Ci	6.47 E-2	8.08 E-2		
iodine-135	Ci	7.53 E-2	6.00 E-2		
Total for period	Ci	1.57 E-1	1.65 E-1		

3. Particulates

strontium-89	Ci	1.40 E-3	1.45 E-3		
strontium-90	Ci	2.83 E-5	1.29 E-5		
cesium-134	Ci	2.30 E-6	2.49 E-6		
cesium-137	Ci	3.31 E-5	4.04 E-5		
barium-lanthanum-140	Ci	4.86 E-3	4.22 E-3		
chromium-51	Ci		1.49 E-5		
manganese-54	Ci	9.19 E-6	1.09 E-5		
	Ci				
	Ci				
cobalt-60	Ci	3.27 E-5	4.11 E-5		
	Ci				
	Ci				
cerium-141	Ci	8.59 E-6	4.66 E-6		
cerium-144	Ci	1.67 E-5	4.10 E-6		
ruthenium-103	Ci	3.19 E-6	1.28 E-6		
ruthenium-106	Ci	4.48 E-5	3.12 E-5		

TABLE 1C

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT (1978)

GASEOUS EFFLUENTS - GROUND LEVEL RELEASE

JULY - DECEMBER 1978

Nuclides Released	Unit	CONTINUOUS MODE Quarter	CONTINUOUS MODE Quarter	BATCH MODE Quarter	BATCH MODE Quarter
1. Fission gases					
krypton-85	Ci				
krypton-85m	Ci				
krypton-87	Ci				
krypton-88	Ci				
xenon-133	Ci		5.49 E+1		
xenon-135	Ci	2.59 E+2	9.57 E+1		
xenon-135m	Ci				
xenon-138	Ci				
Total for period	Ci	2.59 E+2	1.51 E+2		
2. Iodines					
iodine-131	Ci	9.67 E-3	1.56 E-2		
iodine-133	Ci	7.08 E-2	1.14 E-1		
iodine-135	Ci	1.32 E-1	1.99 E-1		
Total for period	Ci	2.12 E-1	3.29 E-1		
3. Particulates					
strontium-89	Ci	6.75 E-4	8.25 E-4		
strontium-90	Ci	3.30 E-6	4.96 E-6		
cesium-134	Ci	2.38 E-5	2.95 E-5		
cesium-137	Ci	9.59 E-5	1.06 E-4		
barium-lanthanum-140	Ci	2.36 E-3	3.99 E-3		
manganese-54	Ci	1.58 E-5	1.17 E-5		
cobalt-58	Ci	2.26 E-6	1.58 E-6		
chromium-51	Ci	5.80 E-5	6.73 E-5		
cobalt-60	Ci	1.33 E-4	8.86 E-5		
zinc-65	Ci		8.58 E-6		
	Ci				
cerium-141	Ci	2.45 E-5	8.85 E-5		
cerium-144	Ci	9.46 E-6			
	Ci				

PILGRIM NUCLEAR POWER STATION

RADIOACTIVE EFFLUENT AND WASTE DISPOSAL REPORT

INCLUDING RADIOLOGICAL IMPACT ON HUMANS

JANUARY 1 THROUGH JUNE 30, 1979

Prepared by: ______________________________
Thomas L. Sowdon
Senior Radiological Engineer

Approved by: Fred J. Mogolesko
Fred J. Mogolesko
Group Leader of Environmental Science Group

Date of Submittal: September 1, 197

TABLE 1A

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT (1979)

GASEOUS EFFLUENTS - SUMMATION OF ALL RELEASES

	Unit	Quarter 1	Quarter 2	Est. Total Error, %
A. Fission and activation gases				
1. Total release	Ci	2.25E+3	3.33E+3	4.00E+1
2. Average release rate for period	μCi/sec	2.89E+2	4.24E+2	
3. Percent of Technical Specification limit	%	1.47E-1	2.01E-1	
B. Iodines				
1. Total iodine-131	Ci	5.07E-2	2.61E-2	3.00E+1
2. Average release rate for period	μCi/sec	6.52E-3	3.32E-3	
3. Percent of Technical Specification limit	%	2.54E0	1.31E0	
C. Particulates				
1. Particulates with half-lives > 8 days	Ci	2.28E-2	< 1.17E-2	3.50E+1
2. Average release rate for period	μCi/sec	2.93E-3	< 1.49E-3	
3. Percent of Technical Specification limit	%	5.40E-1	< 2.13-1	
4. Gross alpha radioactivity	Ci	< 5.12E-7	< 4.25E-7	
D. Tritium				
1. Total release	Ci	5.61E+1	1.84E+1	5.00E+1
2. Average release rate for period	μCi/sec	7.22E0	2.34E0	
3. Percent of Technical Specification limit	%	NA	NA	

TABLE 1B
EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT (1979)
GASEOUS EFFLUENTS – ELEVATED RELEASE

		CONTINUOUS MODE		BATCH MODE	
Nuclides Released	Unit	Quarter	Quarter	Quarter	Quarter

1. Fission gases

krypton-85	Ci	2.51E-2	1.82E-2	———	———
krypton-85m	Ci	4.51E+2	5.01E+2	———	———
krypton-87	Ci	1.18E+1	7.30E+2	———	———
krypton-88	Ci	5.75E+2	1.19E+3	———	———
xenon-133	Ci	4.25E+2	3.00E+2	———	———
xenon-135	Ci	2.66E+1	3.72E+1	———	———
xenon-135m	Ci	7.06E+1	4.26E+1	———	———
xenon-138	Ci	2.84E+2	1.28E+2	———	———
xenon-131m	Ci				
xenon-137	Ci				
xenon-133m	Ci				
Total for period	Ci	1.84E+3	2.93E+3	———	———

2. Iodines

iodine-131	Ci	8.37E-3	2.14E-2	———	———
iodine-133	Ci	3.38E-2	2.88E-2	———	———
iodine-135	Ci	3.98E-2	2.84E-2	———	———
Total for period	Ci	8.20E-2	7.86E-2	———	———

3. Particulates

strontium-89	Ci	1.31E-3	1.58E-3	———	———
strontium-90	Ci	6.69E-6	8.82E-6	———	———
cesium-134	Ci	1.31E-6	7.13E-7	———	———
cesium-137	Ci	3.17E-5	4.20E-5	———	———
barium-lanthanum-140	Ci	3.83E-3	3.25E-3	———	———
chromium-51	Ci				
manganese-54	Ci	4.85E-6	7.19E-6	———	———
cobalt-58	Ci	1.75E-6	———	———	———
iron-59	Ci				
cobalt-60	Ci	2.10E-5	2.11E-5	———	———
zinc-65	Ci				
zirconium-niobium-95	Ci	———	7.54E-7	———	———
cerium-141	Ci	2.08E-6	2.66E-6	———	———
cerium-144	Ci	5.32E-5	3.96E-5	———	———
ruthenium-103	Ci	1.84E-6	———	———	———
ruthenium-106	Ci	1.02E-4	1.58E-4	———	———

TABLE 1C

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT (1979)

GASEOUS EFFLUENTS - GROUND LEVEL RELEASE

		CONTINUOUS MODE		BATCH MODE	
Nuclides Released	**Unit**	**Quarter**	**Quarter**	**Quarter**	**Quarter**
1. Fission gases					
krypton-85	Ci	< 3.67E-5	< 4.10E-5	—	—
krypton-85m	Ci	1.33E+1	1.48E+1	—	—
krypton-87	Ci	1.38E+2	3.86E+1	—	—
krypton-88	Ci	6.61E+1	4.16E+1	—	—
xenon-133	Ci	2.85E+1	2.77E+1	—	—
xenon-135	Ci	1.61E+2	7.64E+1	—	—
xenon-135m	Ci		3.27E+1	—	—
xenon-138	Ci		1.72E+2	—	—
Total for period	Ci	4.07E+2	4.04E+2	—	—
2. Iodines					
iodine-131	Ci	4.23E-2	4.69E-3	—	—
iodine-133	Ci	3.05E-1	2.69E-2	—	—
iodine-135	Ci	6.19E-1	4.56E-2	—	—
Total for period	Ci	9.66E-1	7.72E-2	—	—
3. Particulates					
strontium-89	Ci	1.47E-3	2.55E-3	—	—
strontium-90	Ci	1.01E-5	5.34E-6	—	—
cesium-134	Ci	2.90E-6	4.89E-6	—	—
cesium-137	Ci	4.29E-5	7.85E-5	—	—
barium-lanthanum-140	Ci	1.46E-2	< 3.87E-3	—	—
manganese-54	Ci	1.92E-5	8.59E-6	—	—
cobalt-58	Ci				
iron-59	Ci				
cobalt-60	Ci	2.46E-4	7.96E-5	—	—
zinc-65	Ci				
zirconium-niobium-95	Ci				
cerium-141	Ci	3.68E-4	4.92E-6	—	—
ruthenium-103	Ci				
ruthenium-106	Ci				

THE UNIVERSITY OF UTAH

Friday, October 23, 1992

David Mulligan
Commissioner
Massachusetts Department of Public Health
150 Tremont Street
Boston, Massachusetts 02111

Dear Commissioner Mulligan,

We respectively submit the report of the review committee for the Southeastern Massachusetts Health Study of leukemia around the Pilgrim power plant. The review committee met three times, held one conference call, and have exchanged numerous faxed copies of material and telephone calls. The conclusions reached by the committee were arrived at by consensus. If you have any questions please feel free to call any member of the committee.

Sincerely yours,

Joseph L. Lyon

Joseph L. Lyon, M.D., M.P.H.

Committee Co-Chairman

Department of Family and Preventive Medicine

50 North Medical Drive
Salt Lake City, Utah 84132
(801) 581-7234